NUCLEAR WEAPONS MATERIAL CONTROL: VERIFICATION OF TRITIUM PRODUCTION LIMITATIONS

by

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ABSTRACT

The limitation of tritium production as part of a nuclear weapons material control agreement would reduce concerns of clandestine stockpiles of fissile nuclear material, and possibly limit the production of certain types of nuclear weapons. A regime for verifying tritium production limits was developed in the thesis.

The basic conclusions of the analysis are that a system to verify tritium production limits at a declared production facility can be produced with existing technology. Such a system would be based on the measurement of lithium depletion and tritium production. Measurement techniques for both lithium and tritium are available and have accuracies of approximately 1%. Moreover, although power and large research reactors can be used to produce significant quantities of tritium, non-production at such reactors can be verified with intensive inspection and sampling at fuel fabrication facilities and reactors. Verification of tritium limits would require intrusiveness that is unprecedented in U.S.-Soviet arms control efforts.

Thesis Supervisor: Marvin Miller
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SCOPE NOTE

The following thesis develops a verification regime for an agreement on the limitation of tritium production. It might be used in developing the basis for a U.S-Soviet agreement on the limitation of special nuclear materials production, or as a starting point for developing IAEA safeguards for tritium production[1].

A verification regime includes the following: (1) verification of tritium production limits at declared military production facilities; (2) verification that tritium is not being produced at other declared facilities such as power reactors; and (3) verification that tritium is not being produced at clandestine facilities. The traditional means of international verification at nuclear power reactors is International Atomic Energy Agency (IAEA) safeguards, while National Technical Means (NTM) of verification is commonly used to search for clandestine facilities in arms control agreements. However, there are no established methods for verifying production limits at declared military production facilities. Thus, the focus of the thesis will be on (1).

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[1] The IAEA does not currently safeguard tritium or tritium production technology.

CHAPTER ONE

U.S.-SOVIET ARMS CONTROL

1.1 Types of Nuclear Arms Control

Nuclear arms control encompasses the following:

-- limitation on numbers and types of deployed warheads and delivery systems;
-- prohibition of specific tests required to develop new weapons;
-- prohibition of nuclear weapons in particular regions (i.e. nuclear free zones);[1]
-- requirements for the exchange of information in potentially dangerous situations (i.e. risk reduction);
-- limitation on the production of non-nuclear weapons components; or
-- limitation on the production of nuclear weapons materials such as plutonium, highly enriched uranium, or tritium.

Each of these approaches has been considered in controlling U.S.-U.S.S.R. arms competition. Although materials production limitation has been discussed extensively, only the first five have been implemented in superpower arms control agreements (See table 1.1).

Each form of arms control has potential national security benefits. Controls on the numbers and types of deployed delivery systems and warheads may reduce the number of deployed weapons that are considered destabilizing. Limits on testing inhibit the development of new weapons that

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[1] J.R. Phillips and J.J. Malanify, "Safeguards Technology Applied to Arms Control and Verification," Journal of the Institute of Nuclear Material Management (Summer 1986): p. 26.

are also considered destabilizing, e.g. MIRVed missiles. Requirements for the exchange of information reduce the risk of inadvertent nuclear conflict. The establishment of nuclear free zones reduces the potential areas of nuclear deployment and conflict. Limitations on the production of non-nuclear weapons components can aid in verification of limitations on types and numbers of nuclear delivery systems. In addition, all types of agreements can reduce financial costs, build confidence between nuclear rivals, and increase each side's ability to predict the military capabilities of its adversaries, and thus reduce the likelihood of an overreaction to military developments and the initiation of an arms race.[2]

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[2] Harold Brown, "Thinking About National Security," Nuclear Strategy, Arms Control and the Future (Boulder, CO: Westview Press, 1985), p. 215.

Name of Agreement[3]	Date Signed	Type of Agreement
ANTARCTICA TREATY	1959	Weapons Location
LIMITED TEST BAN TREATY	1963	Test Limitation
HOT LINE AGREEMENT	1963	Risk Reduction
OUTER SPACE TREATY	1967	Weapons Location
TREATY OF TLATELOLCO	1967	Weapons Location
SEA BED TREATY	1971	Weapons Location
NUCLEAR ACCIDENTS AGREEMENT	1971	Risk Reduction
STRATEGIC ARMS LIMITATION TREATY I (SALT I)	1972	Numbers and Types/ Test Limitation[4]
HIGH SEAS AGREEMENT	1972	Risk Reduction
NUCLEAR WAR PREVENTION AGREEMENT	1973	Risk Reduction
THRESHOLD NUCLEAR[5] TEST BAN TREATY	1974	Test Limitation
STRATEGIC ARMS LIMITATION TREATY II (SALT II)	1979	Numbers and Types
INTERMEDIATE AND SHORT RANGE MISSILE TREATY (INF)	1987	Numbers and Types/ Limitation on Non-Nuclear Components

U.S.-Soviet Arms Control Agreements Categorization
Table 1-1

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[3] Dietrich Schroeer, Science Technology and the Nuclear Arms Race (New York: John Wiley and Sons, 1984), pp. 348, 349.

[4] SALT I includes an interim agreement on offensive weapons, as well as strict limits on the development and deployment of defensive weapons (ABM Treaty).

[5] SALT II and the Threshold Nuclear Test Ban Treaty have not been ratified by the U.S. Senate.

Limitations on the production and stockpiling of nuclear weapons material can place an upper limit on the total number of nuclear weapons available to each power, as well as affect the types of nuclear weapons produced. Moreover, such limitations can build confidence in the fairness of international regime established to limit the horizontal spread of nuclear weapons.[6] This regime is centered on the Nuclear Non-Proliferation Treaty (NPT) and International Atomic Energy Agency (IAEA) safeguards. By acceding to the NPT and accepting safeguards, nations agree not to use nuclear materials for military purposes. Although all nuclear powers[7] support the goals of the NPT, they continue to produce nuclear materials for military applications. By halting, or at least limiting the production of nuclear weapons material and accepting some form of safeguards on their nuclear facilities, these nations would have a better chance of convincing skeptical non-nuclear nations to accept IAEA safeguards.

1.2 Nuclear Material Limitation History

From the outset of the nuclear weapons age, nuclear material control has been proposed as a form of arms control.

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[6] The spread of nuclear weapons to nations that previously did not posses nuclear weapons is referred to as "horizontal", while the increase in nuclear weapons stockpiles of nations that already have nuclear weapons is referred to as "vertical".

[7] United States, U.S.S.R., France, China, and U.K.

On June 14, 1946, the U.S. Representative on the Atomic Energy Commission of the United Nations (UNAEC),[8] Bernard Baruch, proposed the creation of an International Atomic Development Authority to control all nuclear materials production.[9] This proposal was based on the fundamental belief that nuclear war could only be averted by the creation of an international organization that would own and manage all fissile material. National ownership of nuclear materials with only international inspection would not, in the eyes of the U.S., provide adequate security. The U.S. plan called for international inspection to detect clandestine plants as well as enforceable sanctions administered by the United Nations.

The U.S., being the only nation in possession of nuclear armaments, further proposed that when the system of international nuclear control was put in place, it would halt manufacture of nuclear weapons, and existing nuclear weapons would be destroyed. In the United Nations Atomic Energy Commission, ten nations approved the U.S. plan, but the Soviet Union abstained. The Soviet Union, fearing that the

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[8] The UNAEC was created in January 1946 by the 51 nations of the United Nations General Assembly to make specific proposals to ensure the peaceful use of nuclear energy. The membership of the UNAEC was limited to the 11 states in the United Nations Security Council and Canada.

[9] "The Baruch Plan: Statement by the United States Representative (Baruch) to the United Nations Atomic Energy Commission, June 14, 1946," Documents on Disarmament 1945-1959, (U.S. Department of State publication no. 7008, 1960), pp. 7-16.

U.S. proposal was merely a means of strengthening and extending the U.S. nuclear monopoly, insisted on concluding a convention on the prohibition of nuclear weapons before international control of nuclear materials was put in place.[10]

Moreover, the Soviets rejected the proposed breadth of the international organization's power as incompatible with state sovereignty. In 1947, Ambassador Gromyko proposed a Soviet plan for nuclear material control. According to the Soviet plan, after U.S. nuclear weapons were destroyed and their use renounced, nationally owned nuclear plants would be supervised by an International Control Commission (ICC). The Commission would ensure peaceful use by periodic inspections and would have access to any facility for "mining, production, and stockpile of atomic raw materials, as well as to the facilities for the exploitation of atomic energy." The ICC would also have the authority to carry out scientific research in nuclear energy to enable it to adequately carry out its inspection responsibility. Yet it would not have the nuclear development purview given to the international agency in the U.S. plan.[11] Although the ICC would have intrusive inspection rights, such rights would not approach the control

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[10] "Statement by the Soviet Representative (Gromyko) to the Security Council, March 5, 1947," Documents on Disarmament 1945-1959, p. 66.

[11] "Statement by the Soviet Representative (Gromyko) to the Security Council, March 5, 1947," p. 70.

implicit in the U.S. proposal (see table 1.2).

The Soviet proposal was rejected by the UNAEC on April 5 1948. A statement endorsed by the majority of the UN concluded that:

> The Soviet Union proposals are not an acceptable basis for international control of atomic energy. The UNAEC cannot endorse any scheme which would not prevent the diversion of atomic material, which provides no effective means for the detection of clandestine activities and which has no provisions for prompt and effective enforcement action. The Soviet Union Government has not only proposed a scheme that is fundamentally inadequate for the control of atomic energy, but at the same time has made the overriding stipulation that they will not agree to establish even such a feeble scheme of control until all atomic weapons have been prohibited and destroyed. It is completely unrealistic to expect any nation to renounce atomic weapons without any assurance that all nations will be prevented from producing them".[12]

In light of these fundamental differences, on May 17, 1948, the United Nations Atomic Energy Commission voted to adjourn indefinitely.[13]

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[12] "Report and Resolution on the Soviet Proposals by the Working Committee of the United Nations Atomic Energy Commission, April 5, 1948," Documents on Disarmament 1945-1959, p. 167.

[13] "Third Report of the United Nations Atomic Energy Commission to the Security Council, May 17, 1948," Documents on Disarmament 1945-1959, p.172.

ISSUE	U.S. Proposal	Soviet Proposal
Date Proposed	June 1946	June 1947[14]
Existing nuclear weapons (U.S.) destruction	after material control put in place	prior to creation of ICC
Ownership/ management of nuclear facilities	international	state
Inspection	N/A	periodic inspection
Search for clandestine nuclear facilities	broad rights	none
Punishment	yes: state can not veto UN sanctions	no: state retains U.N. veto rights
Control of nuclear development	responsible for all development of potentially dangerous nuclear technologies	responsible for only nuclear development needed to improve inspection capabilities

Comparison of U.S. and Soviet Proposals for Nuclear Control
Table 1-2

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[14] The Soviet convention prohibiting nuclear weapons was proposed in 1946.

As both U.S. and Soviet nuclear weapons stockpiles grew and nuclear weapons became entrenched in domestic and international politics, the U.S. requirement that international ownership and management form the basis of a nuclear material control agreement subsided. A military fissile material[15] production cutoff of national nuclear facilities became the next feasible alternative. In 1956 President Eisenhower wrote to Chairman Bulganin:

> ...the United States would be prepared to work out, with other nations, suitable and safeguarded arrangements so that future production of fissionable materials anywhere in the world would no longer be used to increase the stockpiles of explosive weapons.[16]

Moreover, between 1956 and 1959 a cutoff was repeatedly proposed by the U.S. in the United Nations, but Soviet responses were generally negative[17], probably reflecting their smaller fissile material stockpile.

However, more recently, in 1982 Soviet Foreign Minister Gromyko stated that the "cessation of production of fissionable materials for manufacturing nuclear weapons"

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[15] fissile material refers to an isotope such as U-235 or Pu-239 that is capable of sustaining a fission chain reaction.

[16] "Letter from President Eisenhower to Soviet Premier (Bulganin), March 1, 1956," Documents on Disarmament 1945-1959, p. 594.

[17] Frank von Hippel, David Albright, and Barbara Levi, "Stopping the Production of Fissile Material for Weapons," Scientific American, 253, no. 3 (September 1985): pp. 40-47.

could be part of the initial stages of nuclear disarmament.[18] Moreover, the U.S. nuclear freeze movement of the early 1980's encompassed the idea of halting the production of nuclear materials.

The context of the early proposals were much different than that of current proposal. In the 1940's, the total world stockpile of nuclear weapons was small. The U.S. weapons stockpile was 2 in 1945; 9 in 1946; 13 in 1947; and 50 in 1948.[19] In comparison, between 1945 to 1986, the U.S. has manufactured nearly 60,000 warheads of 71 different types.[20] Soviet production history is probably similar.[21] Today, large stockpiles of nuclear weapons materials exist in both countries, and the possibility of a sudden breakout from a nuclear material control agreement with a clandestine stock of fissile material must be considered.

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[18] Frank von Hippel et al, Scientific American, pp. 40,41.

[19] Thomas Cochran, William Arkin, Robert Norris, and Milton Hoenig, Nuclear Weapons Databook Volume II U.S. Nuclear Warhead Production, (Cambridge, MA :Ballinger Publishing Co., 1987), p. 15.

[20] Many of these weapons have been retired. The exact number of nuclear weapons in the U.S. nuclear stockpile is not available. However, several estimates of the stockpile near 25,000 weapons can be found. For example, Cochran et al has estimated that the stockpile in 1982 was 26,000 weapons in "The U.S. Nuclear Stockpile," Arms Control Today, April 1982, p. 1. (Cited by Frank von Hippel in "Soviet Diversion of Plutonium Under IAEA Safeguards, " The Nuclear Weapons Freeze and Arms Control, p. 38).

[21] Moreover, currently, France, England, and China each have substantial overt nuclear stockpiles and other nations are suspected of having clandestine nuclear weapons production capabilities.

1.3 Tritium Control

Although most proposals for nuclear material control have been centered on fissile material used in nuclear weapons, (i.e. plutonium and highly enriched uranium), the additional control of tritium production could supplement a materials control agreement. Because many nuclear weapons use tritium[22], which decays with a 12.32 year half life, the problem of undeclared/undetectable stockpiles could be mitigated by limiting tritium production. According to Herman Roser, Assistant Secretary of Energy for Defense programs, "in the event we were to quit producing tritium this year, the yield of the stockpile would drop....". Although weapons might be developed that do not use tritium, such development would require testing, and thus minimize the potential for breakout.

Moreover, tritium limitations might serve to constrain the deployment of certain types of nuclear weapons. According to Cochran et al, enhanced radiation (ER) weapons utilize tritium and deuterium to produce lethal high energy neutrons. Such weapons are not required for strategic nuclear deterrence. In addition, Cochran claims that ER warheads are used in battlefield nuclear missions as well as

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[22] It is not publicly known which nuclear weapons require tritium, but it is commonly held in the public literature that most or all do. For example, in "The Tritium Factor as a Forcing Function in Nuclear Arms Reduction Talks,"(Science, September, 2, 1988, pp. 1166-1169), J. Carson Mark et al claim that "Tritium represents the key to the compact and efficient designs of modern nuclear weapons."

for anti-ballistic missile systems.[23] Tritium limitations could help to limit both of these types of deployment.

It is probable that an agreement on tritium production would allow some production to continue. Thus, methods would have to be developed that would allow each state to ensure that tritium production limits are obeyed. The following analysis develops a verification regime for an agreement on the limitation of tritium production.

1.4 Verification: An Introduction to Intrusiveness

Verification of an arms control agreement includes both cooperative and unilateral measures that help to ensure that treaty partners are abiding by the agreement's provisions. Different treaties provide for different degrees of intrusiveness for verification. Intrusiveness, in turn, refers to the degree and frequency of access an inspector or inspection mechanism has to a facility, complex, or area. Arms control negotiators must struggle to find a balance between the security benefits of greater verification, and the political difficulties of greater intrusiveness, both domestically and with the negotiating partner.

The least intrusive form of verification is no access. In this case an agreement is verified entirely by National

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[23] Cochran, Nuclear Weapons Databook Volume II U.S. Nuclear Warhead Production, p.23.

Technical Means of Verification (NTM)[24] and no provisions are made for aiding verification. The next level of intrusiveness is verification completely by NTM, but with non-intrusive cooperative measures. This includes designation measures, transparency measures, or collateral measures to assist NTM in verifying treaty provisions. These measures designate the location and function of certain weapons, increase the visibility of certain weapons to NTM, and help to establish a "firebreak" between permitted and non-permitted activities.[25] For example, SALT II designates specific ICBM test ranges for both sides. Moreover, the treaty requires air launched cruise missiles(ALCM)-equipped bombers to have Functionally Related Observable differences from non ALCM-equipped bombers. The ABM treaty, SALT agreements, and the INF treaty contain stipulations that neither side will attempt to impede verification by NTM. The INF agreement further stipulates that the U.S.S.R will open the roofs of SS-25 missile garages on demand six times per year to assist NTM. Non-intrusive cooperative measures can include the exchange of data or blueprints of military

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[24] NTM refers to a broad range of systems for collecting intelligence, including reconnaissance satellites, ships and aircraft used to monitor Soviet missile tests, and ground stations, such as the large phased array radar on Shemya Island, Alaska. (source: Message From the President of the United States Transmitting The Treaty Between the U.S. of America and the U.S.S.P. On the Elimination of Their Intermediate-Range and Shorter-Range Missiles, p.24.)

[25] William F. Rowell, Arms Control Verification, (Cambridge, MA: Ballinger Publishing Co., 1986), p. 56.

facilities.[26]

The most intrusive form of verification is direct on site inspections (OSI). OSI is intended to allow treaty partners access to needed information that can not be gained with NTM and other cooperative measures, and can also build confidence between treaty partners. With direct OSI, treaty parties are allowed access to specific sites, facilities, or areas in the host country to inspect areas or remote monitoring devices. Within the realm of OSI is a continuum of possible degrees of intrusiveness. Possible forms of OSI include:

-- continuous inspection presence;
-- intermittent, announced inspections;
-- scheduled inspection;
-- unannounced inspections;
-- unattended monitoring
-- perimeter monitoring; and
-- remote flow monitoring.[27]

Each of these forms of inspection can further be sub-divided by access allowed during inspection. The INF treaty is the only concluded U.S.-Soviet treaty that provides for OSI.

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[26] An example of non-intrusive cooperative measure that was never enacted is President Eisenhower's 1955 "open skies" proposal, whereby the U.S. and U.S.S.R. would allow each other the right to aerial reconnaissance in order to reduce fears on both sides of missile deficiencies. The proposal included the exchange of blueprints of military establishments to further facilitate verification.

[27] "Verification of Arms Controls on US and Soviet Fissionable Materials," EPRI RP-620-50, draft (further reference not available).

1.4.1 On-Site Verification: a short record

The degree of intrusiveness called for in the INF treaty is unprecedented in U.S.-Soviet arms control negotiations as it calls for fairly comprehensive on-site verification. Although provisions for on-site verification have been prominent in negotiations for a comprehensive nuclear test ban treaty, such a treaty has not been concluded.
The INF treaty stipulates several forms of OSI. This includes baseline inspections to verify data exchanges; closeout inspections to verify elimination of specified facilities; elimination inspections to verify the completion of the process of elimination with respect to items lost or accidentally destroyed, placed on static display, or with respect to training equipment; short notice inspections of specific facilities to ensure that treaty-prohibited activities have ceased; and continuous portal or perimeter verification to ensure that certain missile components are not clandestinely produced.

Agreement	Date Signed	Treaty Verification[28] Provisions
LIMITED TEST BAN TREATY	1963	None[29]
STRATEGIC ARMS LIMITATION TREATY I (SALT I) (ABM)	1972	1,2,3,5
THRESHOLD NUCLEAR TEST BAN TREATY[30]	1974	1
STRATEGIC ARMS LIMITATION TREATY II (SALT II)	1979	1,2,3,4,5,6
INTERMEDIATE AND SHORT RANGE MISSILE TREATY (INF)	1987	1,2,3,4,6,7

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1) NTM
2) Prohibition of Concealment Measures
3) Designation Measures: designating location and function of certain weapons.
4) Transparency Measures: increase visibility of systems to NTM.
5) Collateral Measures: designed to cut off most likely routes of evasion; establish a "firebreak" between permitted and non-permitted activities.
6) Data exchange
7) OSI

Cooperative Measures in Arms Control Agreements
Table 1-3

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[28] Information in table derived from Rowell, pp. 55-59.

[29] The Limited Test Ban Treaty specifically omitted any reference to NTM.

[30] The Threshold Nuclear Test Ban and SALT II have not been ratified by the U.S. Senate.

CHAPTER TWO

IAEA SAFEGUARDS AS A MODEL FOR OSI

Although the INF Treaty provides for the most intrusive verification of any U.S.-Soviet arms control agreement, the most intrusive and widely applied multi-national arms control verification occurs in the form of International Atomic Energy Agency (IAEA) safeguards. These safeguards are the technical and procedural means by which the IAEA verifies the political obligation undertaken by many states not to use peaceful nuclear facilities for military purposes. Although there are substantial differences between IAEA safeguards and U.S.-Soviet arms control verification, much of the basic technology and experience used and gained by the IAEA can be applied to superpower arms control.

2.1 I.A.E.A. Safeguards Development

The creation of an organization to both promote and safeguard the peaceful uses of nuclear energy grew out of President Eisenhower's 1953 "Atoms for Peace" plan, but had its conceptual origin in the previously discussed Soviet proposal of 1947. The Soviet proposal called for the creation of an International Control Commission with limited periodic inspection rights to verify non-military use of state-controlled nuclear facilities. Although President Truman had earlier rejected this approach as incapable of

preserving security, in 1953 President Eisenhower proposed to the United Nations' General Assembly that an international agency be established to allocate nuclear material, and to safeguard the peaceful uses of supplied material. In the President's plan, this agency would become the main promoter of international nuclear development. The initial planning of the agency's statute contained provisions for its right to impose safeguards to assure that supplied material would not be misused.[1]

On October 20, 1956, the International Atomic Energy Agency statute was opened for signature. The fundamental safeguard provision required the Agency to "ensure, so far as it is able, that assistance provided by it or at its request or under its supervision or control is not used in such a way as to further military purpose".[2] However, the statute provided only a very general framework for safeguards application.[3]

The Agency's first detailed safeguards system was described in 1961 in "The Agency Safeguards System"

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[1] Safeguards Against Nuclear Proliferation, Stockholm International Peace Research Institute, (Cambridge, MA: The MIT Press, Cambridge, MA, 1975), p. 4.

[2] "Statute of the International Atomic Energy Agency," Article II, paragraph 1.

[3] "Statute of the International Atomic Energy Agency," Article XII.

(INFCIRC/26)[4]. However, the procedures in INFCIRC/26 were only designed for reactors with less than 100 megawatt thermal power, and not for larger reactors or other fuel cycle facilities. In 1964, procedures were adopted for larger reactors (INFCIRC/26/Add. 1): In 1965, the Agency adopted a revised system (INFCIRC/66), which was further defined and extended in 1966 (INFCIRC/66/Rev.1), with the inclusion of reprocessing plants, and in 1968 to cover conversion and fabrication plants (INFCIRC/66/Rev.2). INFCIRC/66/Rev. 2 is the basis of a large number of current safeguard agreements.[5]

2.2 Safeguards Under INFCIRC/153

Safeguards application was further defined and standardized with INFCIRC/153. This document forms the model for application of Agency safeguards in countries that have acceded to the Treaty on the Non-Proliferation of Nuclear Weapons, and thereby agreed to place all domestic nuclear facilities under safeguards. INFCIRC/66 and earlier safeguards documents apply only to specific facilities. INFCIRC/153 describes specific verification procedures, goals and obligations.

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[4] An IAEA INFormation CIRCular is typically identified by its INFCIRC number, e.g. INFCIRC/26.

[5] Safeguards Against Nuclear Proliferation, p.6.

2.2.1 Safeguards Objective:

INFCIRC/153 defines the objective of safeguards as the timely detection of diversion of significant quantities of nuclear materials from peaceful uses to the development of nuclear explosives.[6] It is assumed that the threat of detection is sufficient to deter diversion.

A "significant quantity" corresponds roughly to the amount of nuclear material with respect to which the possibility of manufacturing a nuclear explosive device can not be excluded. For plutonium a significant quantity is 8 kilograms and for uranium enriched to greater than 20% U-235 a significant quantity is 25 kilograms.[7] Detection is "timely" if it occurs before the diverted material can be used to fabricate a nuclear weapon. Nuclear material is defined as any source of special fissile material; Tritium is not safeguarded.

2.2.2 IAEA Safeguards Implementation:

IAEA safeguards under INFCIRC/153 are based on an overlapping system of state accounting and reporting, with Agency verification of state reporting by application of independent measurements, inspection, and containment and surveillance measures.

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[6] INFCIRC/153, paragraph 28.

[7] "The Present Status of IAEA Safeguards on Nuclear Fuel Cycle Facilities," IAEA Bulletin 22, no. 3/4 (August, 1980): pp. 4,5.

2.2.2.1 Developing Safeguards Specifics

Specific detailed safeguards provisions must be developed for each safeguarded facility. These provisions are specified in the Subsidiary Arrangements for each facility. In the development of the subsidiary arrangements, the state provides the Agency with certain design information. This includes:

> -- identification of the facility, including its general character, purpose, capacity, and location;
> -- a description of the general arrangement of the facility with reference to the form, location and flow of nuclear materials, and to the general layout of important items of equipment which produce or process nuclear material;
> -- a description of the features of the facility relating to material accountancy, containment and surveillance; and
> -- a description of procedures for nuclear material accountancy and control.[8]
>
> With this design information, the Agency determines:
>
> -- material balance areas (MBAs)[9] for verification,
> -- key measurement points (KMPs);
> -- the timing and procedures for taking physical inventory;
> -- records and reporting requirements;
> -- requirements and procedures for verifying the location and quantity of nuclear materials; and
> -- containment and surveillance methods, techniques, and points of application.

In order to verify this design information, the Agency may, in co-operation with the state, conduct on-site inspections.

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[8] INFCIRC/153, paragraph 43.

[9] An MBA refers to an area in or outside a facility such that the quantity of nuclear material in each transfer into or out of it can be determined; and the quantity of nuclear material in it can be determined when required for safeguards purposes. (source: INFCIRC/153, paragraph 110.)

During such inspections, the Agency inspector has access to any location where the initial report indicates nuclear material is present.[10]

2.2.3 Routine and Special Agency Inspections

In addition to the above described inspection, the Agency conducts routine and special inspections. Routine inspections consume the largest part of inspection effort.[11] Activities during such Inspections include:

-- the examination of records;
-- independent measurements at KMPs;
-- verification of functioning and calibration of measurement equipment;
-- application and use of surveillance and containment measures.[12]

These activities are intended to verify that reports sent by the state to the agency are consistent with facility operator records[13]; verify location, quantity and composition of all safeguarded material; and verify causes for lost material and shipper/receiver differences. During routine inspections, the Agency has access only to the strategic points (KMPs) and

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[10] INFCIRC/153, paragraphs 46,48,76.

[11] F. Kilk, "Field experience of safeguards inspectors," IAEA Bulletin, 23, no. 4 (Dec. 1981): p. 15.

[12] INFCIRC/153, paragraph 72.

[13] INFCIRC/153 requires that the state provide the Agency with reports of inventory changes within 30 days of the end of the month in which the change occurs, and material balance reports showing the material balance based on a physical inventory of nuclear material in each MBA. (paragraphs 62,63,64,65)

records specified in the Subsidiary Arrangements.[14]

Special inspections are made, in addition to routine inspections, when the Agency believes that information provided by the state is not adequate for the Agency to fulfill its responsibility, or when an unusual incident or circumstance leads the state to believe that a loss of nuclear material in excess of that specified in the subsidiary arrangement may have occurred. A special inspection may involve Agency access to information or locations that it does not have in its routine inspections.[15] Disagreements on Agency access for special inspections are settled by arbitration.[16] The Agency informs the state of inspection results as well as the conclusions it has drawn from its verification activities.

The number, intensity, duration, and timing of routine inspections are kept to a minimum, consistent with the effective implementation of safeguards procedures, and therefore vary by facility. A tension exists between the desire on the part of the safeguards community for greater confidence in detecting possible diversions of nuclear material and the interests of both the plant operators and the state. The former do not want international inspectors to interfere with plant operation, or to obtain proprietary

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[14] INFCIRC/153, paragraph 76.

[15] INFCIRC/153, paragraph 73.

[16] INFCIRC 153, paragraph 22.

information. The latter are concerned with ceding state sovereignty to an international organization. For these reasons IAEA safeguards are designed to avoid hampering peaceful economic and technological development; to avoid "undue" interference with nuclear activities and plant operation; to carry out safeguards practices in a safe manner; and to protect commercial and industrial secrets.[17]

2.3 Safeguards Devices

The devices utilized by the IAEA in fulfilling its safeguarding obligation fall into two categories: those used for Containment and Surveillance (C/S), and those used for Material Accountancy (MA).[18]

2.3.1 Containment/Surveillance (C/S)

C/S devices include optical cameras, seals, and radiation detectors. These devices are used to verify that the containment of nuclear material has not been breached, and that nuclear material has left locations only by legitimate routes. These devices greatly reduce the manpower requirements of the Agency inspection effort.

Generic requirements for C/S devices are that the device must be:

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[17] INFCIRC/153, paragraphs 4,5.

[18] According to INFCIRC/153, paragraph 29, "the (safeguards) agreement should provide for the use of material accountancy as a safeguards measure of fundamental importance, with containment and surveillance as important complementary measures."

-- reliable;
-- capable of unattended operation over a long period of time;
-- tamper proof/tamper indicating;
-- easy to service;
-- capable of operating with little maintenance; and
-- capable of operating without interference with plant operation.[19]

These same requirements apply to U.S.-U.S.S.R arms control verification.

2.3.2 Analytic Techniques and Methods

Because the Agency does not have the resources to independently measure all items, and the plant operator might object that such measurements would be unduly intrusive, the Agency verifies material accountancy data supplied by the state by making independent measurements on a randomly sampled subset of items.[20] These measurements may include both destructive chemical analysis and non-destructive analysis. The former measurements are usually very accurate but expensive, and results are typically available only after three to five weeks, since samples must be sent to the Agency laboratory for analysis. The latter measurements are less accurate, but results are often available immediately.[21] The Agency takes a large number of the low accuracy measurements

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[19] "Surveillance and Containment Measures to Support IAEA Safeguards," IAEA Bulletin 19, no. 5 (Oct. 1977): p. 21.

[20] "IAEA Safeguards Technical Manual," report number IAEA-174.

[21] A. von Baekman, "The application of modern methods and techniques in safeguards operations," IAEA Bulletin 23, no. 1 (Dec. 1981): p.15.

and a relatively small number of the high accuracy measurements.

2.4 Safeguards Applied to U.S.-Soviet Arms Control

Many of the techniques and technology developed for the IAEA can be applied to U.S-Soviet arms control verification. In fact, as discussed in section 2.1, IAEA safeguards are conceptually based on the 1947 Soviet proposal for superpower nuclear material control. However, several significant differences exist between verification requirements of a superpower arms control agreement and those of current safeguards application. These differences are discussed below and summarized in table 2-1.

IAEA methods were developed for application in civilian facilities. The focus of the current analysis is the application of safeguards to military production facilities. This conceptual difference manifests itself in both the development of the safeguards approach and in defining the objectives of safeguards.

Approach

In developing an IAEA safeguards approach, designers must ensure that safeguards do not unduly affect the safe and efficient operation of the facility. In developing safeguards for military facilities, however, the safeguards designer must consider not only safety and efficiency factors, but also national security concerns of the inspected

party. The inspected party can legitimately oppose inspector access to a location in a military facility because such access would divulge information that would be directly detrimental to its national security interests. In the case of application to peaceful facilities, the inspected state can not legitimately oppose access based on national security concerns, because such facilities are not supposed to have a direct national security significance.

Goal

The goal of safeguards application is altered by the military/civilian dichotomy. The goal of IAEA safeguards is to deter by timely detection the diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear explosives. The goal of the verification regime is to deter not only the diversion of nuclear material, but also the production of excess nuclear material.

Because the IAEA monitors only peaceful facilities, it has no right to stipulate what type or rate of nuclear production the state undertakes. In the present application, however, the goal is not simply to monitor the flow of nuclear material, but also to ensure that production does not exceed an agreed limit. The agreement provisions in the IAEA case are violated only when material is diverted. In the present context, agreement provisions are violated when material is diverted, or production exceeds agreed

quantities.

Significant Quantity

Because tritium is not required for a basic fission explosive and is considered useful in nuclear weapons program only after a country has already developed basic fission devices, the Agency does not safeguard tritium. Thus, there is no defined "significant quantity" of tritium as there is for plutonium, uranium and other nuclear materials.

Moreover, the determination of the composition of a significant quantity in the context of U.S.-Soviet arms control is much different than that in the context of IAEA safeguards. In the case of IAEA safeguards, a significant quantity is roughly that which is required to construct one nuclear weapon. Since both the U.S. and Soviet Union have many thousands of nuclear weapons, diversion of one nuclear weapon's worth of nuclear material is not significant. In the application of safeguards to arms control, a significant quantity is the amount of material required to make a strategically or politically significant difference in superpower arsenals, and is thus variable. This issue is explored quantitatively in chapter four.

Removal From Safeguards

In the case of IAEA safeguards, material is removed from safeguards only after it has been consumed or diluted; has become irrecoverable; or if the state wishes to use the

nuclear material in non-nuclear activities.[22] In the U.S.-Soviet context, the goal is primarily to monitor production: After tritium for weapons is produced and accounted for, it would no longer be safeguarded.

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[22] INFCIRC/153, paragraph 11.

ISSUE	U.S.-Soviet	IAEA-INFCIRC/153
Guiding Document	Treaty	INFCIRC/153
IAEA Technology	applicable, but not currently used	currently used by definition
Flexibility of approach	limited only by domestic politics and national security concerns	must be consistent with relevant INFCIRC and Agency Statute
Facility type	military/civilian	civilian
Search/ Investigative Ability	yes: NTM and cooperative verification provisions	none
Police Search Power	no	no
Punishment/ Sanctions/ Coercive measures	withdraw from treaty, trade agreements, etc./ violence	inform U.N.
Reasons for limiting inspector access	national security concerns and safe efficient plant operation	safe and efficient plant operation/ state sovereignty
Materials Verified	fissile material/ tritium	fissile material
Significant Quantity of Material	depends on strategic conditions	corresponds to one nuclear weapon
Inspection Agency	U.S/U.S.S.R. verification agency	IAEA multinational

Comparison of I.A.E.A. Safeguards and U.S.-Soviet Verification
Table. 2-1

CHAPTER THREE

TRITIUM PRODUCTION

3.1 Understanding the Cycle

In order to adequately define a verification regime for a nuclear material, the production cycle and physical characteristics of that material must be understood. In the case of plutonium and uranium which are safeguarded by the IAEA, generic fuel cycles are well known. In the case of tritium, however, production information is often tightly controlled by the governments producing the material. Nevertheless, a great deal of information is available through declassified U.S. government safety reports; declassified information on alternative production processes which have been developed and used by the U.S. in the past; unclassified information on the French tritium production efforts; and unclassified information on tritium processing for the U.S. fusion energy program. Since the goal of this analysis is to develop a generic verification regime to be applied to both the U.S. and Soviet programs, as well as possibly allied programs, and since the international diffusion of technical information makes similarities in various production processes likely, borrowing details from different tritium production programs does not detract from the analysis.

3.2 Physical Characterization of Tritium

Tritium is a radioactive isotope of hydrogen, consisting of one proton, one electron, and two neutrons. It decays to helium-3 by beta decay with a 12.32 year half life.

$$H^3 \text{ ---> } He^3 + e$$

In this decay, the maximum beta energy is 18.6 KeV, and the average beta energy is 5.7 KeV. The specific activity of tritium is 9600 Ci/g and the specific power is .3240 w/g.[1]

The primary radiologic hazard from tritium is ingestion, since its low energy beta will not penetrate the skin. The maximum Permissible Body Burden recommended by the ICRP is 1 millicurie. The body will assimilate tritiated water and distribute it throughout the body. Tritiated water has a 10.5 day biological half life. The median lethal dose for tritium is 10 Ci, but higher doses can be tolerated with increased fluid intake.[2]

3.2.1 Production Overview

Tritium is very rare in nature. Its natural abundance is 10^{-17} that of hydrogen. The primary man-made sources of tritium are military production reactors and commercial power

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[1] John R. Bartlit, "Hydrogen Isotope Processing in Fusion Power Applications," p.22.

[2] Encyclopedia of Chemical Engineering Technology, Vol. 7, (Kirk-Othmer, John Wiley & Sons,1979), p.561.

reactors.[3] In production reactors, tritium is produced intentionally by the irradiation of Li6 targets.

In light water power reactors, the primary method of tritium formation is ternary fission in which the fissile isotope breaks into two heavy isotopes and tritium. This occurs in 1 out of 10^4 fissions.[4] Estimated production of tritium by ternary fission in a 1000 MWe plant is about 11,200 Ci/yr, assuming an 80% reactor capacity factor.[5] Tritium is also produced by neutron reactions with lithium contaminants in the fuel and boron in control rods and coolant[6], as well as by the irradiation of small quantities of deuterium in the light water. Tritium production in the coolant of a 1000 MWe PWR is only approximately 550 Ci/yr.[7] while estimated tritium production in BWR control rods is

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[3] Considerable quantities of tritium have been released into the environment by atmospheric detonations of thermonuclear explosive devices. Estimates range from .7 kg to 5 kg of tritium released into the atmosphere per megaton-equivalent explosion. (Source: Delmar L. Crowson, "Man-Made Tritium," Tritium, p. 26.)

[4] Samual Glasstone and Alaxander Sesonske, Nuclear Reactor Engineering (third edition), (New York: Van Nostrand Reinhold Company 1981), p. 592.

[5] Assuming .33 thermal efficiency: 3x10^9 J/S x 3.1x10^10 fissions/J x 3.15x10^7 s/yr x .8 capacity factor x 1x10^-4 tritium atoms/fission x 3/6.02x10^23 grams/atom x 9600 Ci/g = 11,212 Ci/yr.

[6] $B^{10} + n^1 \longrightarrow Be^8 + H^3 + 12$ MeV (fast reaction)
$B^{11} + n^1 \longrightarrow Be^9 + H^3 + 9.6$ MeV

[7] Mason Benedict, Thomas Pigford and Hans Wolfgang Levi, Nuclear Chemical Engineering (New York: McGraw Hill, 1981), pp. 391-394.

60,000 Ci/yr.[8]

In heavy water moderated power reactors, the primary method of tritium formation is the neutron bombardment of deuterium. The following reaction occurs with a 2200 m/sec cross section of 5.7x10^-4 barn.

$$H^2 + n^1 \rightarrow H^3$$

For a 1000 MWe CANDU-type power plant, the yearly production rate of tritium in the heavy water is approximately 1.9x10^6 Ci/yr (200 grams per year) at 80% capacity.[9]

In military production reactors, tritium is produced by the reaction of slow neutrons with Li^6 targets. The following reaction occurs with a 2200 m/sec cross section of 940 barn.

$$Li^6 + n^1 \rightarrow He^4 + H^3 + 4.78 \text{ MeV}$$

3.2.2 Tritium Uses

Current tritium use in the U.S. is dominated by the nuclear weapons program. D-T fusion provides a copious source of 14 MeV neutrons and is characterized by a reaction threshold temperature of about 10 KeV.[10]

$$H^2 + H^3 \rightarrow He^4 + n^1 + 17.6 \text{ MeV}$$

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[8] Herbert Kouts, "Tritium Production in Nuclear Reactors," Tritium, p. 40.

[9] Marvin Miller, "Technology to Extract Tritium from Heavy Water," Department of Nuclear Engineering, M.I.T., October 1987 (revised), p. 2.

[10] Bartlit, p. 22.

Nevertheless, tritium is not defined as Special Nuclear Material by the Department of Energy, nor safeguarded by the IAEA, since it is not required for a basic fission explosive.

Official information on tritium production rates in the U.S. are not available. However, Cochran, et al, have estimated that at the end of fiscal year 1984, the U.S. tritium stockpile was approximately 80 kilograms, and that approximately 10.2 kg was produced that year[11]. Annual production rates vary depending on strategic development plans. For example, in 1984, the tritium required to sustain the estimated 80 kg stockpile was only 4.4 kg per year.[12] Although the amount of tritium used in each nuclear weapon is not available, an order of magnitude estimate can be gained by dividing the 80 kg estimated total stockpile by the 25,000 weapons the U.S. is often assumed to have in its stockpile.[13] This calculation yields 3.2 grams of tritium per weapon, assuming all weapons use tritium.

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[11] Thomas Cochran, William Arkin, Robert Norris, and Milton Hoenig, Nuclear Weapons Databook Volume II, U.S. Nuclear Warhead Production, (Cambridge, MA: Ballinger Publishing Co., 1987), p. 180.

[12] The annual tritium required to replace that lost due to decay is 1-exp(-.693/12.32)=5.47%.

[13] The exact number of nuclear weapons in the U.S. nuclear stockpile is not available. However, several estimates of the stockpile near 25,000 weapons can be found. For example, Cochran et al has estimated that the stockpile in 1982 was 26,000 weapons in "The U.S. Nuclear Stockpile," Arms Control Today, April 1982, p. 1. (Cited by Frank von Hippel in "Soviet Diversion of Plutonium Under IAEA Safeguards," The Nuclear Weapons Freeze and Arms Control, p. 38).

The same characteristics of the D-T fusion reaction that make tritium desirable for weapons make it desirable for fusion power applications. Tritium is considered to be the most likely fuel for thermonuclear energy producing reactors because its threshold reaction temperature is lower than that of other possible fusion reactions, e.g. about 10 KeV for D-T fusion versus 50 KeV for D-D fusion.

Considerable quantities of tritium will be consumed and produced in such reactors.[14] A 1,200 MWe reactor of current Tokamak design will burn 536 grams of tritium per day, and breed 562 grams per day.[15] It will require an initial supply of approximately 10 kg, which will subsequently be recovered from excess tritium produced in the blanket, and will have approximately an 11.6 kg tritium inventory. The inventory will be primarily retained in the reactor's breeder blanket, in storage in the form of uranium tritides, and in the tritium processing and blanket tritium recovery systems.[16] However, the future of fusion power is not certain, and

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[14] In the fusion reactors of current Tokamak design, tritium is bred in a blanket of lithium surrounding the D-T plasma, where fusion occurs.

[15] A fusion reactor can produce more tritium than it consumes by utilizing the reaction n+Li7 ---> T + He4 + n - 2.47 MeV. Although this reaction consumes energy, it produces both a tritium atom and a neutron. The produced neutron can subsequently absorbed by Li6 to produce a second tritium atom.

[16] "Starfire-A Commercial Tokamak Fusion Power Plant Study," report number ANL/FPP-80-1, Sept 1980, p. 14-26.

current research only requires tens of grams per year[17].

The primary commercial use of tritium is as a power source for phosphorescent lighted watches, telephone lights, exit signs etc.. In 1986, 62 grams of tritium were supplied by the U.S. Department of Energy to U.S. companies and research institutes, and 117 grams were supplied to foreign companies and research institutes. An additional 1.2 grams were supplied for unspecified Department of Energy projects.[18]

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[17] Cochran, Nuclear Weapons Databook Volume II, U.S. Nuclear Warhead Production, p. 77.

[18] "List of DOE Radioisotope Customers with Summary of Radioisotope Shipments, FY 1986," report number PNL-5948, 1987, p. 5.4.

Atomic Symbol	H3
Molecular Weight	6.032 g per mole
Radiation	B^-
max. energy	18.67 KeV
average energy	5.7 KeV
Decay Formula	$H^3 \rightarrow He^3 + e$
Common Form	gas (HT) or tritiated water (HTO)
gas density at std. cond.	0.2691 g/L
Production Reaction	$Li^6 + n \rightarrow He^4 + H^3 + 4.8$ MeV
Estimated U.S. Stockpile	80 kg [19]
Estimated annual U.S. production	varies by year
Fusion Threshold of DT reaction	10 KeV
Specific Activity	9600 Ci/g
Primary Uses	nuclear weapons; power source for phosphorescent lighting,
Physical Half life	12.32 years
Biological Half life	10.5 day[20]
Cost	$1.35/Ci[21]

Table 3-1 Summary of Physical Characteristics of Tritium

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[19] Cochran, Nuclear Weapons Databook Volume II, U.S. Nuclear Warhead Production, p. 180.

[20] Encyclopedia of Chemical Engineering Technology, Vol. 7, p.561.

[21] Correspondence from Anne Nowicke, Savannah River Laboratory, August 8, 1988.

3.3 Tritium Production Cycle

Tritium is currently produced in the U.S. by the irradiation of enriched lithium^{6} targets and control rods in three heavy water moderated/cooled Department of Energy production reactors at Savannah River, South Carolina. The tritium production process for weapons has seven steps potentially relevant for safeguards efforts: (1) lithium Enrichment[22]; (2) fabrication of enriched uranium fuel and lithium targets; (3) target irradiation; (4) tritium extraction; (5) tritium purification; (6) tritium isotopic enrichment; and (7) tritium storage.[23] The entire production cycle is shown diagrammatically in Figure 3-1.

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[22] Natural lithium contains 7.48% Li^{6} and 92.52% Li^{7}. (Source: S Villani, Isotope Separation, ANS, 1976, p.339).

[23] In this analysis, lithium mining and milling are not considered relevant for safeguards as the amount of lithium on the earth's surface is so great as to make efforts to verify its extraction impractical. The U.S. is estimated to have 320,000 tons of proven reserves, and 4,600,000 tons of land based resources. In addition, the average concentration of lithium in water is .17 g/m^{3}. (Source: "The Fuel Cycle of Fusion Reactors," in Engineering Aspects of Fusion Research, G. Casini, 1980, p. 393).

It should be noted that uranium ore is also not safeguarded by the IAEA.

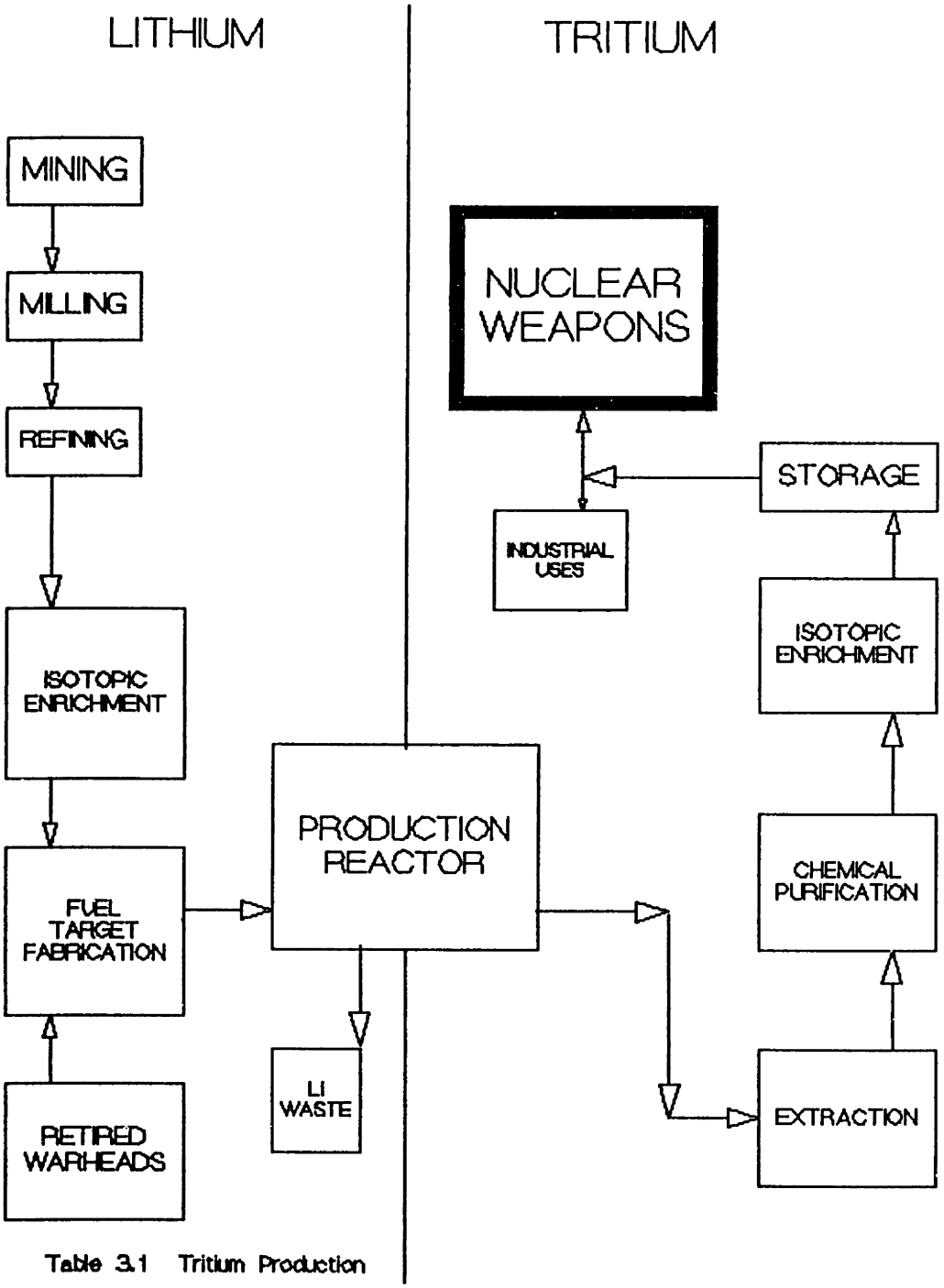

Table 3.1 Tritium Production

3.3.1 Lithium enrichment:

As mentioned earlier, tritium is easily produced through the irradiation of Li^6 targets. Since naturally occurring lithium contains only about 7.5% Li^6, tritium production is most efficiently achieved if the lithium is enriched in Li^6. The enrichment level of lithium used in U.S. tritium production is not publicly available, however in the French tritium production program, the lithium is enriched to 95% Li^6.[24] For the purpose of this analysis, we will assume the U.S. program does the same.

Tested methods for enriching lithium on a small scale include ion exchange, electrolysis, chemical exchange, and ion migration.[25] In the U.S., lithium was enriched in the Y-12 plant at Oak Ridge National Laboratory using a chemical exchange process. In this process separation is achieved by exchange between lithium amalgam and an aqueous solution of lithium hydroxide. Li^6 is concentrated in the aqueous phase which consists of LiOH equal to its limit of solubility[26]. The separation factor for this process is 1.06 to 1.07.[27]

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[24] P. Hugony, H. Sauvage, and E. Roth, "Tritium Production in France," report number ERDA-tr-286, translated from Bulletin d'Information Scientifiqe et Technique, No. 178, Feb 1973, p. 3.

[25] Benedict, p. 641.

[26] Eiche Saito and Gregoire Dirian "Process for the Isotopic Enrichment of Lithium by Chemical Exchange," U. K. patent 902,755, August 9, 1962, p.2.

[27] Benedict, p. 801.

In the United States, enriched lithium is currently obtained from existing stockpiles or retired thermonuclear warheads, which also utilize enriched lithium. The U.S. lithium enrichment facility is currently on standby because "sufficient lithium inventory currently exists."[28] Moreover, in its 1985 budget request, the Department of Energy requested funds to decommission the facility.[29]

3.3.2 Target Fabrication

Tritium can be produced from targets made of lithium metal, alloy, ceramic, or salt.[30] Currently in the U.S. and France, lithium aluminum alloy is used as target material. Previously, tritium was produced at the Hanford nuclear reactors using lithium aluminate ceramic and lithium fluoride.[31] Extruded target tubes must be produced, and the lithium compound must be formed to fit into the target tubes.

The Savannah River reactors are designed to produce

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[28] HAC, FY 1983 EWDA, Part 4, p.257, cited in Thomas Cochran William Arkin, Robert Norris, and Milton Hoenig, Nuclear Weapons Databook, Volume III U.S. Nuclear Warhead Facility Profiles, (Cambridge, MA: Ballinger Publishing Co., 1987), p. 75.

[29] HAC, FY 1985 EWDA, Part 4, p.334, cited in Cochran, Nuclear Weapons Databook Volume III, U.S. Nuclear Warhead Facility Profiles, p.75.

[30] D.G. Jacobs, Sources of Tritium and its Behavior Upon Release to the Environment, U.S. Atomic Energy Commission, 1968., p. 23.

[31] The first targets irradiated at the Hanford site were made of lithium fluoride. Lithium aluminum and aluminate can be irradiated for a longer period of time and at higher flux density than lithium fluoride.

principally tritium and plutonium for weapons, although other radioisotopes have been produced.[32] Plutonium is typically produced in Mark 15, Mark 16B, and Mark 31 assemblies, but tritium is produced in Mark 22 fuel/target assemblies, Mark 16B inner targets, Mark 60B blanket assemblies, and in control rods (lithium is the standard absorber for SRP control rods).[33] As shown in Figure 3-2, the lithium charges can either be an integral part of the fuel (Mark 16B and 22), or a separate target (Mark 60B). Moreover, different targets contain different amounts of Li^6 and therefore produce different quantities of tritium.

Detailed information on fuel dimensions and lithium densities is not available. However, simple calculations show that each Mark 22 assembly contains roughly 66 grams of enriched lithium.[34]

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[32] John P. Church et al, "Safety Analysis of Savannah River Production Reactor Operation (Deleted Version)", Savannah River Laboratory report number DPSTSA-100-1, September 1983, p. 1-1. The "other" isotopes produced are not specified.

[33] Church, pp. 4-12 - 4-18.

[34] A rule of thumb for tritium production is that a reactor designed for tritium production can produce a roughly 1/80 grams of tritium per MWDth of reactor operation. This rule is used in calculations throughout this analysis.

The rule is supported by the fact that a natural uranium fueled, light-water cooled, graphite moderated reactor will produce roughly 1 gram of plutonium in the fuel per MWDth of energy produced (source: Nuclear Proliferation Factbook, pp. 554-555, J.R. LaMarsh). If, in place of the natural uranium fuel, highly enriched uranium fuel is used, neutrons that had been captured in U238 in the fuel can instead irradiate lithium targets surrounding the fuel. In this case, roughly

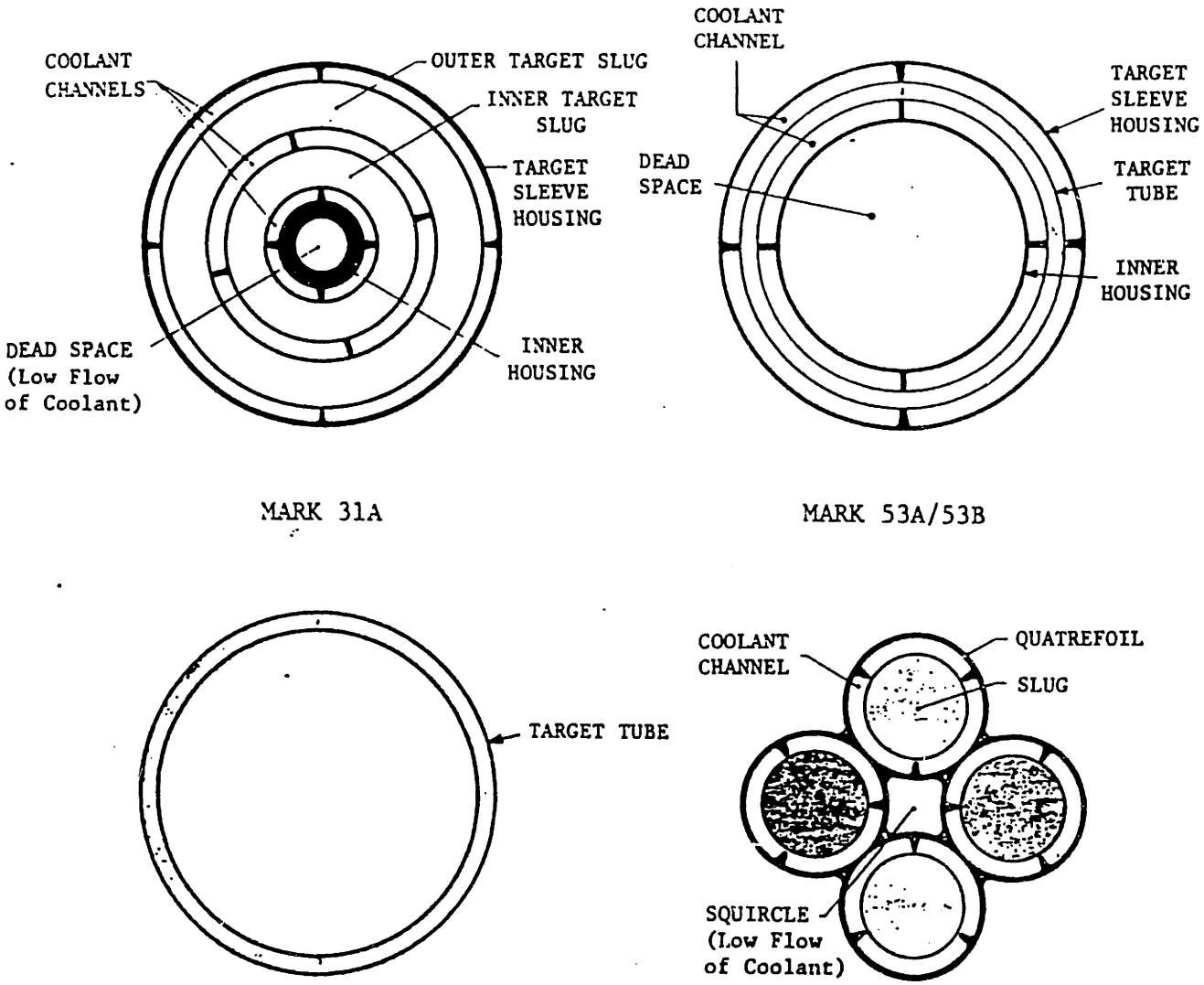

Figure 3-2
Current Target Designs

Source: Church, p. 4-18.

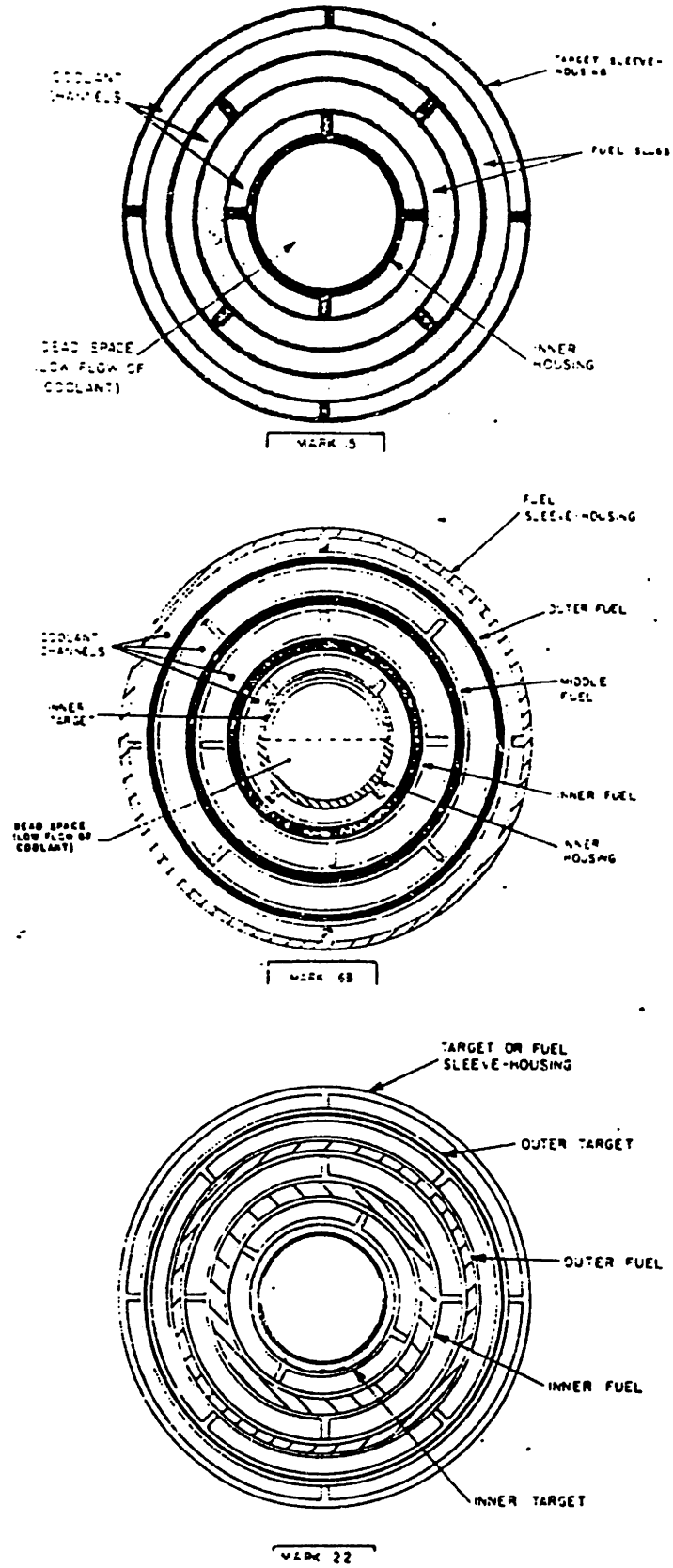

Figure 3-2
Current Fuel Designs

Source: Church, p. 4-17

3.3.3 Target irradiation

Target/Fuel assemblies are received in the assembly area, irradiated in the reactor, and transferred to the disassembly area (Figure 3-3).

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the same number of moles of tritium will be produced as plutonium had previously been produced. Thus, the maximum tritium production rate would be roughly (3/239)x(1 g/MWD). The exact production potential of a given reactor would have to be determined from a detailed analysis of that reactor. The SRL reactors, for example, use fuel that is only enriched to 70-80% U235, which would decrease tritium production. However, they are cooled (and moderated) by heavy water, which would increase tritium production, since fewer neutrons would be absorbed in the coolant.

Since a core of Mark 22 assemblies has a discharge burnup in the range of 400,000 to 500,000 MWth-Day (see section 3.3.3.3) and such a core consists of 400 assemblies, tritium production in each assembly is between 12.5 and 15.625 grams.

Assuming lithium is enriched to 95% (French program), and can only be depleted to 50 % due to buckling (see section 3.4.5), fuel assemblies must contain roughly 66 g of enriched lithium.

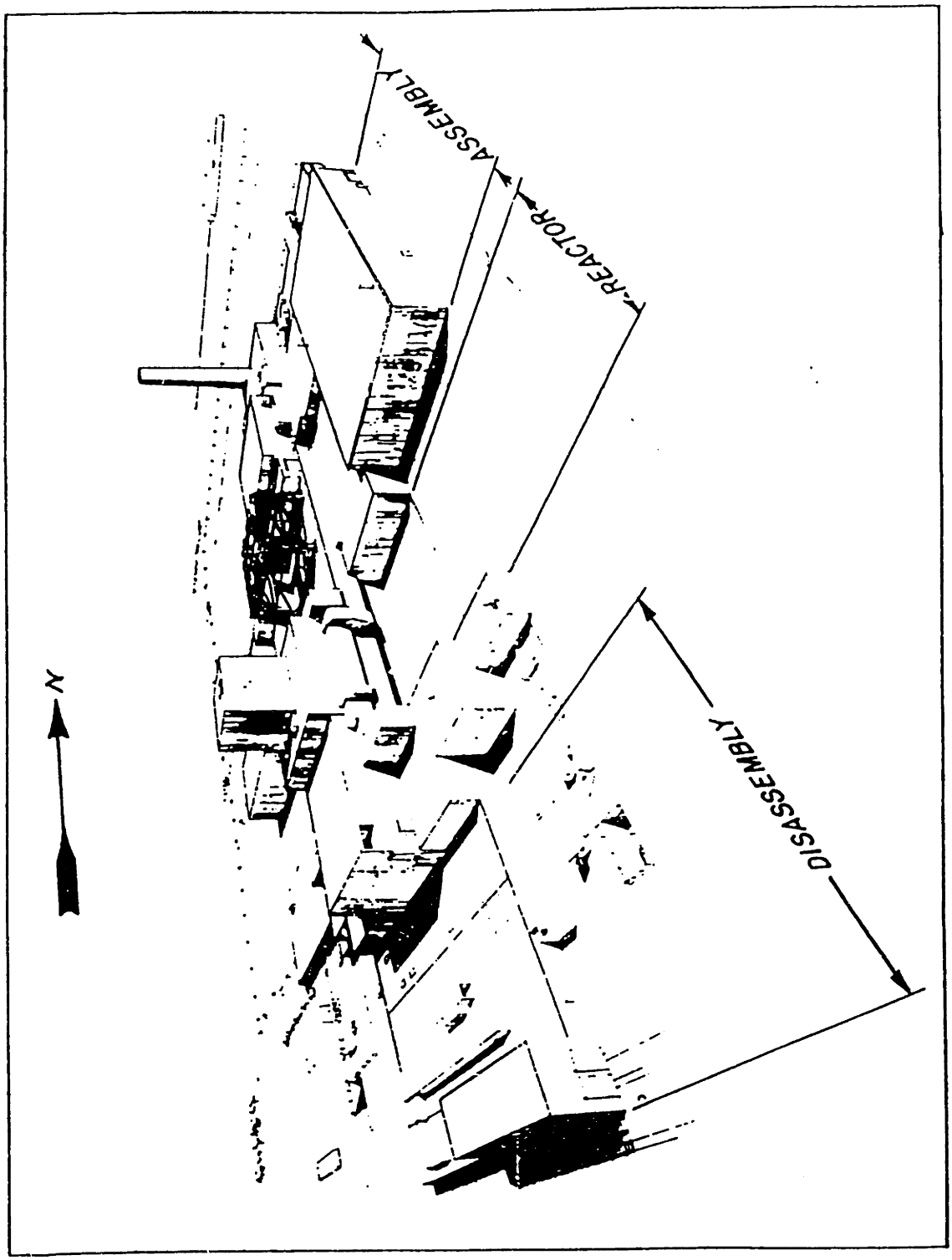

Figure 3-3
Layout of SRP Facility for Tritium Production

Source: Halkard Mackey, Jr. (compiler), "Environmental Information Docket L-Reactor Reactivation," Savannah River Lab Report # DPST-81-241, p.3-3.

3.3.3.1 Assembly Area

Targets and fuel are received and stored in the assembly area of the reactor building. A storage area large enough to store one and one-half full charges of assembled components is located there. Racks and hangars are used to maintain sufficient distance between assemblies to avoid criticality. Material is transferred from this area to the transfer station in the reactor room through a shielded slot in the process room wall.[35]

All components are charged and discharged using remotely operated cranes (Figure 3-4). The charge machine and discharge machine, shown below, are essentially identical. Both machines have three masts to move equipment vertically in and out of the core. The primary difference between the charge and discharge machine is that the discharge machine has a mast to provide water to cool irradiated components.[36] Charge and discharge operations are usually done with an automatic sequence control using a pre-punched control tape, but the control panel can also be operated manually.[37]

In order to reload the reactor, it must be shut down and the upper plug and plenum must be removed and replaced.

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[35] Halkard Mackey, Jr. (Compiler), "Environmental Information Document L-Reactor Reactivation," Savannah River Laboratory report number DPST-81-241, p 9-1.

[36] Church, p.9-4.

[37] Mackey, p. 3-18.

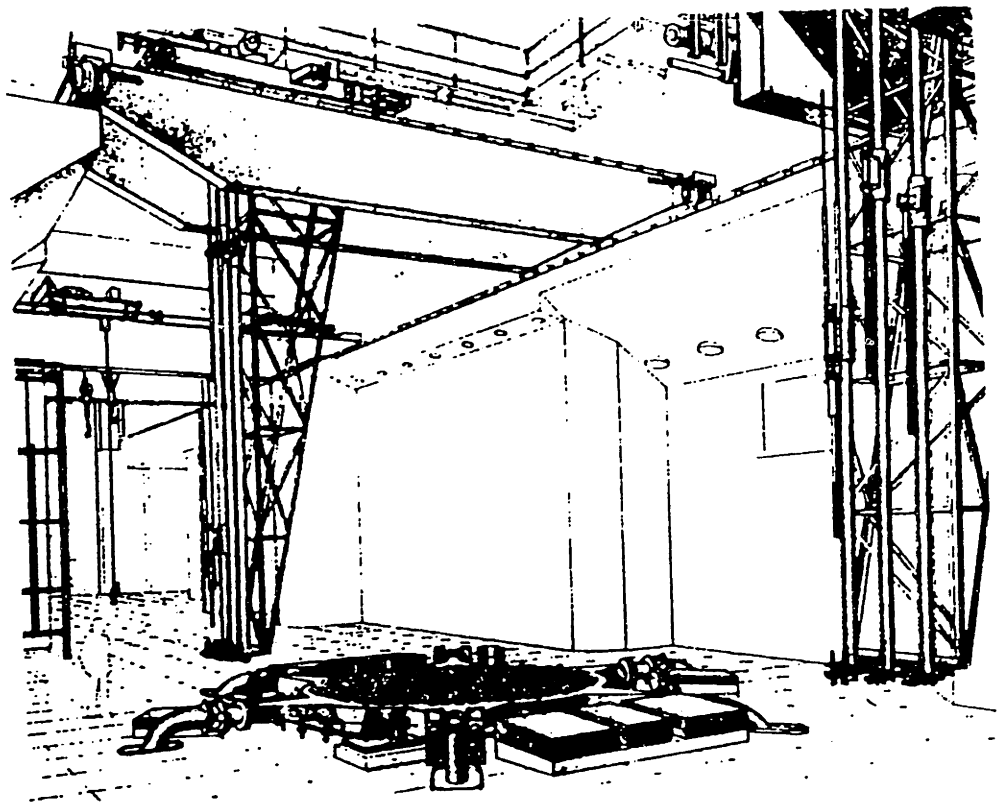

Figure 3-4
Charge and Discharge Machines in Reactor Room

Source: Mackey, p. 3-19

3.3.3.2 Reactor Design

The reactor tank (Figure 3-5) is 15 ft high, and has an inside diameter of 16.25 feet. It is made of 0.5 inch thick type 304 stainless steel plate and has six effluent nozzles spaced around its circumference. The top and bottom shields are 18.75 feet in diameter and 3.33 feet thick. The water plenum is a hollow disk weldment 17 feet in diameter and 9.75 feet thick with six tapered inlet nozzles around its circumference.[38] The reactor must be off and the water plenum and top shield removed before fuel can be reloaded.

A schematic of the reactor area for the P,L and K reactors is shown in Figure 3-6, and the core lattice arrangement is shown in Figure 3-7. Since the reactors began operation in 1953 to 1955, they have operated with over 20 different lattice designs.[39] The core has positions for approximately 600 fuel/target assemblies.[40]

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[38] J.W. Joseph, Jr. and R.C. Thornberry (Compilers), "Analysis of the Savannah River Reactor Emergency Core Cooling System," Savannah River Laboratory report number, p. DPST-70-463, p. A1.

[39] Joseph, p.9.

[40] Joseph, p. A3.

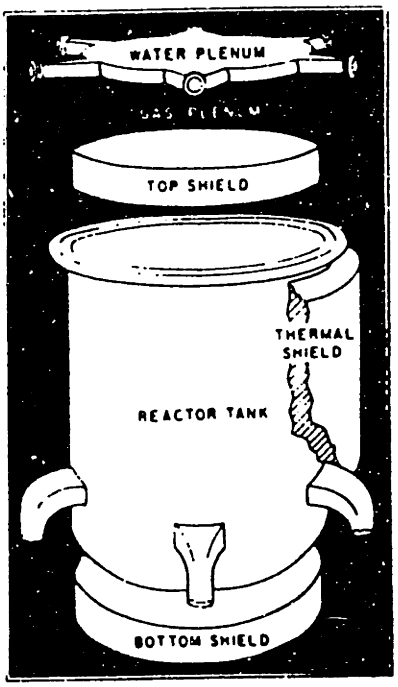

Figure 3-5
Reactor Structure (Schematic)

Source: Mackey, P. 3-12

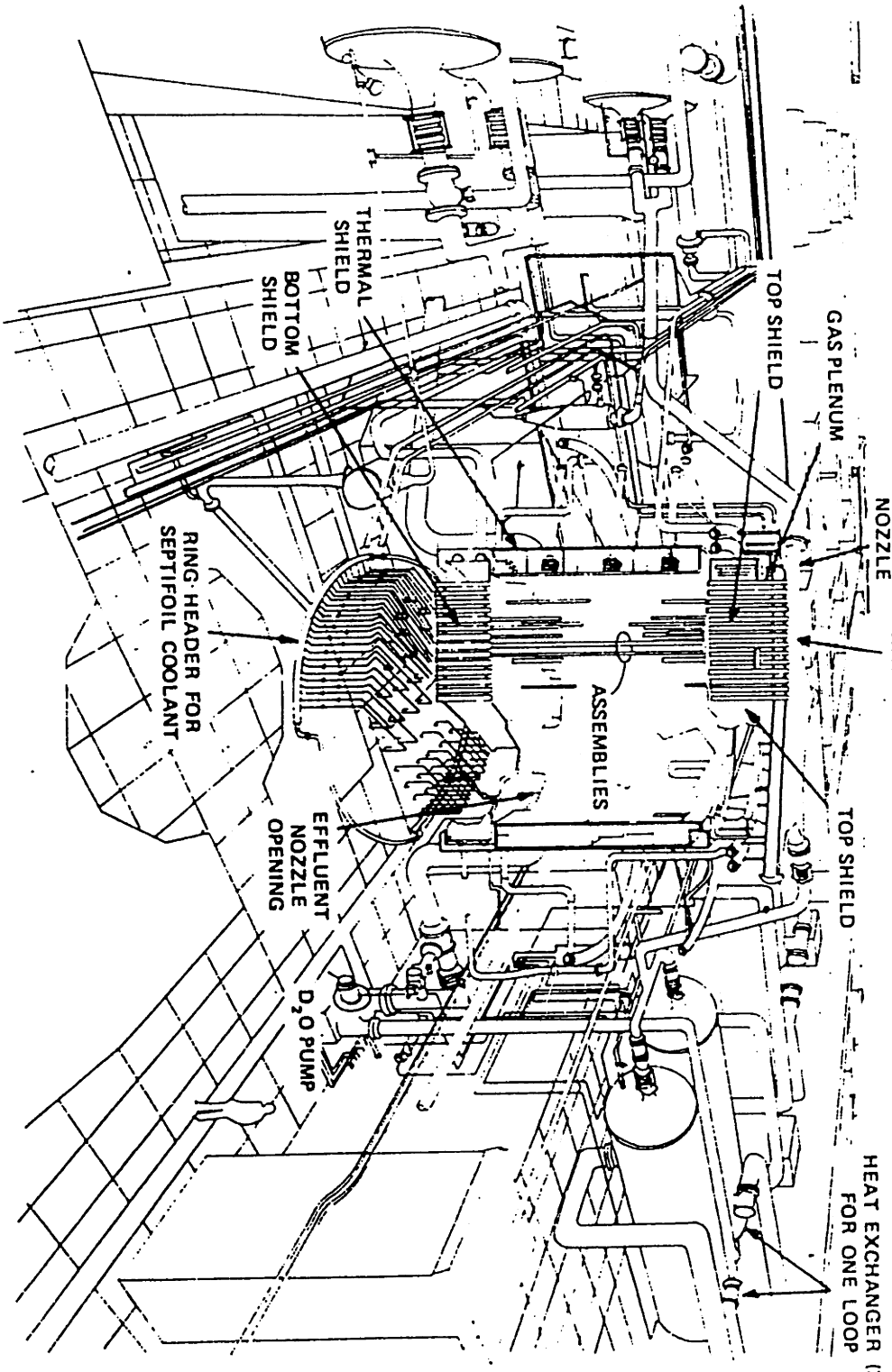

Figure 3-6
Structure of P, L, K Reactors

Source: Mackey, P. 4-3

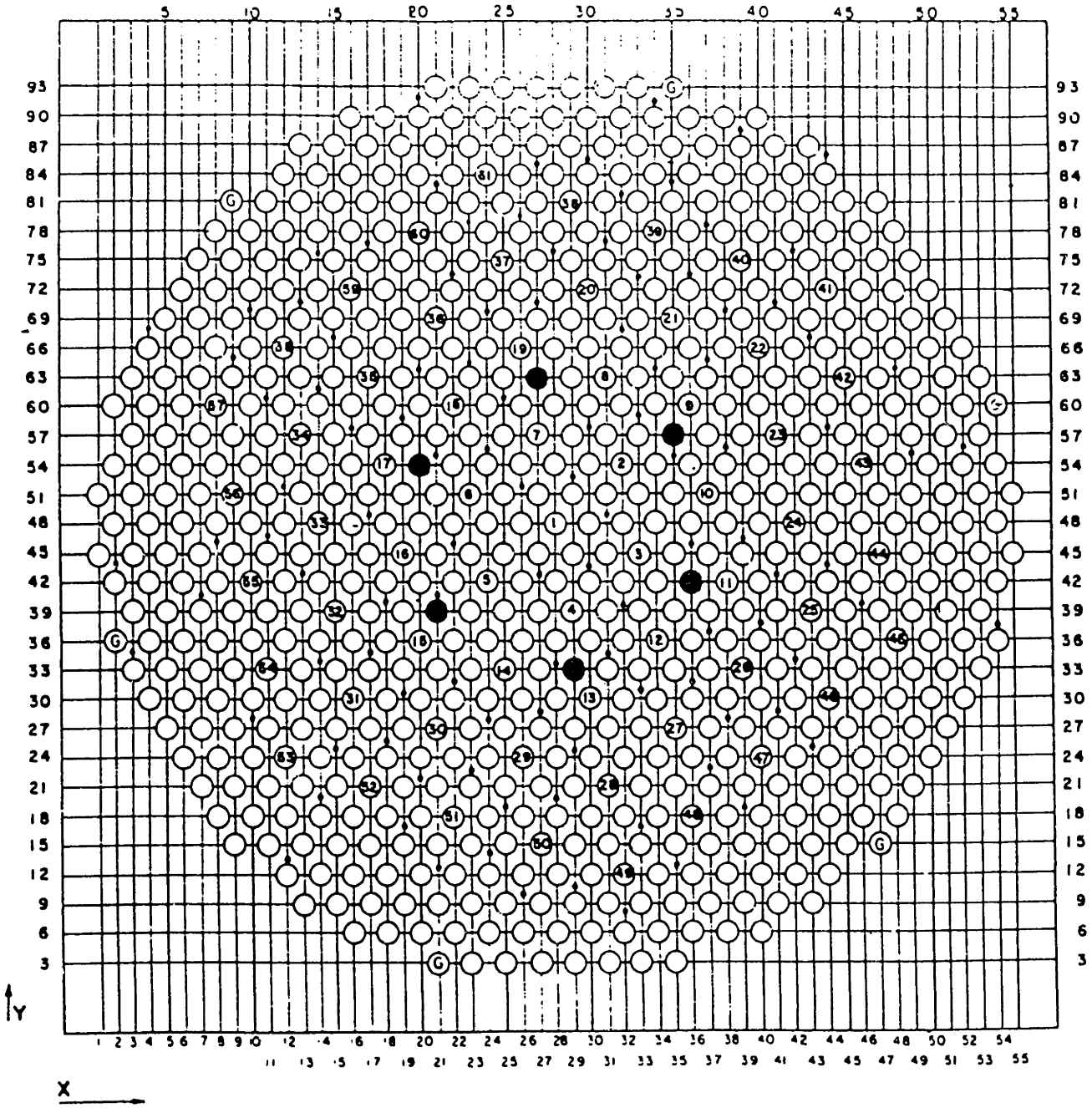

Figure 3-7
Lattice Arrangement for P, L, K Reactors

Source: Mackey, D. 4-7

3.3.3.3 Target Exposure

Savannah River reactors have operated with irradiation cycle lengths varying from 4 to 400 days with a thermal neutron flux that has varied from 5x10^13 to 7x10^15 n/cm2-s.[41] Although longer exposure of the lithium targets leads to greater tritium production, target exposure must be limited to prevent dimensional instability, blistering, creep collapse of cladding, and loss of tritium. These effects occur when the number of tritium atoms is greater than the number of lithium atoms.[42] Thus maximum irradiation occurs when about half of the Li^6 atoms are consumed.

According to Savannah River Laboratory's safety reports, the burnup of Mark 22 fuel is in the range of 400,000 to 500,000 MWD per core loading [43], which corresponds to an irradiation cycle length of 166 to 208 days at 2400 MW power. Currently one reactor has been dedicated solely to tritium production.[44] Such runs are probably loaded with a uniform core of Mark 22 assemblies [45] consisting of enriched Li and

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[41] Church, p. 4-22.

[42] Church, p. 4-47.

[43] Church, p. 4-26.

[44] Cochran, Nuclear Weapons Databook Volume II, U.S. Nuclear Warhead Production, p.60.

[45] According to Church, p. 4-16, although some tritium is produced in control rods, Mark 60B assemblies, and Mark 16B targets, "Much larger tritium production rates are achieved with Mark 22 fuel, loaded in a uniform charge."

highly enriched uranium.[46]

3.3.3.4 Fuel/Target Disassembly

After irradiation, target/fuel is transferred to the disassembly area by a hangar and monorail system for storage, disassembly, and subsequent shipment to the separation facilities (tritium extraction, purification, etc.). This area contains equipment for transfer, storage, disassembly and examination of irradiated material. Its basin is a large concrete-lined pool of water.[47] Assemblies are stored in the basin in a vertical position until decay heat is low enough for further operations.

During discharge, assemblies are suspended in air: Water drainage is collected in a drip pan which swings into place below the assembly once the assembly is clear of reactor. D_2O is also flushed from the assembly while in transit by H_2O flowing from the discharge mast. The water is pumped to drains for storage.

3.3.4 Tritium Recovery:

Tritium is separated from irradiated lithium-aluminum targets, purified and packaged. Targets are extracted from

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[46] According to Cochran, Nuclear Weapons Databook Volume II, U.S. Nuclear Warhead Production, p. 71, fuel for SRL reactors comes from spent fuel returns from the nuclear Navy and U.S. and foreign research and test reactors. The estimated post irradiation enrichment of the Naval fuel is 78%, and the DOE accepts spent research reactor fuel with a remaining enrichment of about 70% U235. Thus, the HEU in SRL drivers is probably between 70 and 80% enriched in U235.

[47] Mackey, pp. 3-1 - 3-7.

assemblies; then, in the French process, rinsed and dried to eliminate alumina charged with heavy water formed during irradiation. Subsequently they are loaded into furnaces where the heat melts the aluminum, and a gas consisting of tritium, helium-4, helium-3, and small quantities of deuterium and protium[48] is released.[49] The hydrogen and helium isotopes are then separated and the hydrogen enriched in tritium. All transfer operations must be done by remote telemanipulation due to activation products, such as cobalt-60, zinc-65, and scandium-46, produced from impurities in the lithium and aluminum.[50]

In France, the irradiated targets are placed in a crucible of stainless steel. The crucible is then introduced into a cell which is placed in a "muffle" oven. Melting takes place in vacuo after pumping out the furnace. Gases are extracted continuously. After a pre-heating period, during which the temperature of the target is brought to 300 C., the temperature is increased to completely melt the target.[51] According to the U.S. patent of the extraction process, the alloy melts at approximately 635 C and the

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[48] Protium is formed by the irradiation of He3, which is formed by the decay of tritium.

[49] E.L. Albenesius and L.H Meyer, "Analytic Techniques for the Use and Control of Tritium at Savannah River," Savannah River Report number DP-771, p. 13.

[50] Hugony et al, p. 3.

[51] Hugony, pp. 3-5.

process takes approximately 1/2 hour for an individual target.[52] After cooling, the crucible is extracted. The released gases are collected in several cylinders connected directly to a line of analytic equipment. After sampling, the gases are transferred to intermediary storage where they will next be purified.

After the gas is extracted, the remaining lithium may be sampled. In France, the sampling of the ingot is carried out by a device which permits the removal of chips and their transfer to the laboratory by pneumatic circuit. In the U.S., however, no such sampling occurs. Moreover, in the U.S., it is not economically efficient to recover the enriched lithium. Spent furnace melts contain up to 70 Ci of tritium per ft^3. The stainless steel furnace crucible containing the spent melt is buried directly at the burial site. Tritium content was established by analysis of samples from the spent melt.[53]

3.3.5 Chemical Purification

After extraction, the mixed gas is purified: isotopes of hydrogen are separated from other gases. Possible purification processes include adsorption on fixed beds at low temperatures, gettering by activated metals, freeze off

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[52] Bernard Abrahamson, "Tritium Production by Neutron Irradiation of Aluminum-Lithium Alloys," U.S. Patent 3,100,184, August 6, 1963, section 3: paragraph 2.

[53] W.J. Jacober, "Tritium Control Technology - Separations," Savannah River Plant report number DPSPU 73-30-7, May 1973, p.19.

on cold surfaces, condensation of hydrogen in the presence of non-condensible helium, and diffusion through palladium membranes.[54] In the U.S. and France, purification is primarily accomplished through diffusion through palladium membrane or hydrogen fixation on pyrophoric uranium.

Diffusion through palladium columns is feasible because hydrogen is the only one of the extracted gasses which diffuses through palladium. In France, the palladium column is made of a stainless steel tube of external diameter 4.8 cm. and of length 77 cm. The column contains 64 diffusion tubes with wall thickness of .076 mm. The gas diffuses outwardly, increasing in enrichment as it passes through each diffusion tube. Each purification requires 20 liters of initial gas.[55]

Purification can also be accomplished by hydrogen fixation on pyrophoric uranium in the form of hydrides. A uranium bed is placed in a vacuum insulated vessel at room temperature. Hydrogen isotopes will chemically bond to uranium, while helium gas is not adsorbed and can be pumped off. The uranium hydride is subsequently decomposed to reconstitute the hydrogen isotopes. [56]

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[54] Bartlit, p. 31.

[55] Hugony, p. 5.

[56] Charles Lindsay, Ronald Sprague, and Jeffrey Brandenburg, "A Measurement Control Study for Tritium Gas," Mound Laboratory report number MLM-3441, July 8, 1987. Lindsay, p. 11-12.

3.3.6 Isotopic Purification

The gas from the chemical purification is nearly 95% tritium, the remainder is composed primarily of the other isotopes of hydrogen. Many processes for the isotopic separation of hydrogen isotopes are known: cryogenic distillation; gas chromatography; palladium chromatography; palladium diffusion; laser separation; electrolysis; and thermal diffusion.[57] Electrolysis is relatively simple for modest enrichments on a small scale. However, for large scale continuous separation processes, thermal diffusion has the advantage, especially if tritium is in the gas phase.[58] Diffusion columns can be simply constructed to provide automatic and continuous separation that yield several liters of tritium per day with purity exceeding 99%.[59] At Savannah River Laboratories, both thermal diffusion columns and cryogenic stills are used.[60]

Thermal diffusion is the process of partial separation of gases due to a temperature gradient. In this process, the heavier isotope moves toward the cooler region.
The French thermal diffusion columns are each made of two

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[57] Bartlit, p.35.

[58] Jacobs, p. 38.

[59] Jacobs, p. 35.

[60] "An In-Line Analyzer for Monitoring Gas Composition In Tritium Purification Process - Design", Philippe Chastagner, p. 119.

annular tubes, 2.5 meters tall, heated by a central wire. The initial gas mixture is confined between the inner heated tube, and the outer tube, which may be cooled. In thermal diffusion columns, convection currents are set up, so that the lighter isotope flows upward and the heavier flows downward, thereby allowing the multiplication of the separation effect. A sketch of a generic thermal diffusion column appears in Figure 3-8. The entire French cascade is composed of four separate diffusion columns. Tritium loss in this process is less than 1% of initial tritium.

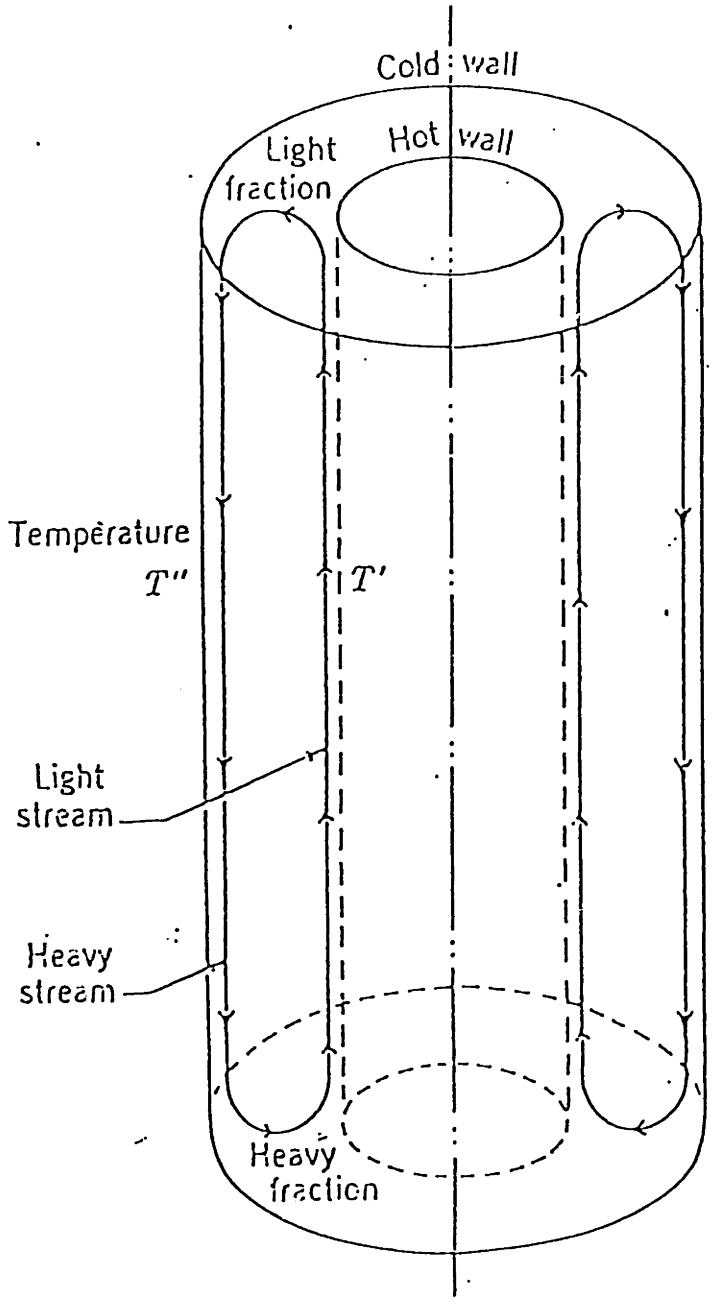

Figure 3-8

Thermal Diffusion Column

3.3.7 Tritium Storage:

After isotopic purification, the tritium is prepared for delivery for both military and civilian applications. Tritium is conveniently stored in metal tritides (e.g. uranium or titanium) or in gas form at ambient temperatures. In the U.S. tritium gas is shipped off plant at low pressure (<2atm). Typical primary containers for tritium are double-walled stainless steel, nickel alloy or tin or tin coated steel and hold 12 or 50 liters of gas.[61] The primary container is sealed within an aluminum vessel, carried in a steel drum. Each primary container is sealed with a closure valve.[62]

Tritium is loaded by connecting the product loading line with a vacuum type rubber sealed compression fitting to the primary container. After closing the manual valve on container, the container is removed from the loading line and is sealed with a vacuum type rubber seal.[63]

Titanium and uranium can also be used to store tritium. Titanium is more stable and favorable for use in longer-term storage. At room temperature, tritium can be stored in the form of UT_3. The tritium can easily be recovered by heating to a temperature near 400 C. Titanium, alternatively, must be heated to over 650 C to recover at 1 atm. The density of

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[61] Bartlit, pp. 40-47.

[62] Jacober, p. 20.

[63] Jacober, p. 20.

UT_3 is 10 g/cm^3 [64] allowing a tritium the storage capacity of 0.36 g/cm3.

3.4 Material Losses

A key consideration in accounting for tritium is the verification of process losses, especially if such losses can exceed a significant quantity (defined in section 4.2). If such losses are large, and difficult to account for, the credibility of the verification regime could be undermined. Although information on the losses in U.S. facilities is not available, relevant information can be gleaned from the French production program.

The global yield (g) of the French tritium production facility is approximately 96%, where

$$g = \frac{\text{recovered } H^3 \text{ } (H^3 \text{ produced} - H^3 \text{ lost})}{Li^6 \text{ consumed and lost in production.}} = 0.96 \text{ [65]}$$

Thus, assuming no lithium is lost in the process, the fraction of tritium lost in the production process is 4%. Since the French tritium production techniques are similar to the U.S., it is reasonable to assume that material loss in U.S. facility is also approximately 4%.

The primary sources of losses are:

-tritium decay in process in the plant;

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[64] Bartlit, pp. 40-47.

[65] Hugony, pp. 3-17.

-diffusion of tritium through hot elements in the installation;
-gas residues remaining in depleted slugs; and
-exposure to air of circuits when serviced.

According to Hugony et al, "An exact account of these diverse items establishes a balance close to 100%". Although this report does not give a further quantitative breakdown of the sources of losses, it can be roughly calculated as follows.

Tritium decay during production is quantified by the fundamental decay rate formula:

$$dn/dt=-Ln+cr$$

where
1-n is the number of tritium atoms decayed during fuel irradiation;
L is the decay constant for tritium (1.54 x10^-4 d^{-1}); and
cr is the conversion rate of lithium to tritium.

Solving this equation yields:

$$n=cr/L(1-exp(-Lt))$$

For an irradiation time on the order of 200 days the fraction lost due to decay is:

$$1-n/ntot=1-1/Lx200\ (1-exp(-200L)=1.5\%.$$

where ntot is the total number of Li6 atoms consumed=Lx200

Thus, tritium decay accounts for 1.5% of missing material. This number can be further refined by allowing decay of total product during processing after irradiation, and this brings decay to 2%. Thus, 1.5%-2% loss of the 4% loss is attributable to tritium decay.

The second source of tritium loss is attributed to

diffusion through hot elements in the process equipment. The only such hot elements are in the thermal diffusion columns, and in the extraction process. Loss in the thermal diffusion columns is probably greater due to greater gas exposure to hot elements. In the French process, the yield of the diffusion process "is better than 99%", suggesting that losses are on the order of 1%.

Gas residues in waste slugs are the third source of tritium loss. As previously mentioned, in the U.S., spent furnace melts have a tritium concentration of 70 Ci/ft3. Mark 22 target/fuel assemblies have a target volume on the order of 1 ft^3.[66] Since each of these elements produce roughly 15 grams of tritium, the tritium density prior to extraction in each is (9600Ci/g x 15 grams)/1 ft^3= 1.4x10^5 Ci/ft^3. Thus, gas residue in waste is a minimal source of tritium loss in the U.S. process (i.e. < .1%).

Exposure to air of pipes during servicing is not as easy to quantify as other forms of tritium loss. However, subtracting the first three sources from the total loss yields losses in the range of 1% to 1.5%. Tritium exposure to air can be monitored by tritium in air concentration monitors, and stack monitors. Thus, in a well run facility,

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[66] This value was calculated based on a assembly length of 14 feet, and target radius based on scaled measurements of sketches of the Mark 22 to sketches of other fuel assemblies for which dimensions were provided. This method is clearly not very accurate, but only an order of magnitude measurement is required.

tritium losses are on the order of 4% of tritium production, and most of this can be accounted for by known mechanisms.

CHAPTER FOUR

SAFEGUARDS REGIME

As discussed earlier, the IAEA does not currently safeguard tritium.[1] Thus, there are no IAEA models on which to base a superpower tritium limitation verification system. The current chapter presents and analyzes a safeguards regime for tritium utilizing the technical analysis in chapter three, and IAEA methodology presented in chapter two.

The basic purpose in developing a tritium safeguards regime is to deter production above agreed limits by making detection of such production likely. As will be shown in section 4.3, in order to do so, it is necessary to monitor both Li6 depletion and tritium extraction.

A safeguards analysis must include production assumptions and quantitative verification objectives, as well as an analysis of possible diversion paths. Moreover, it must include an analysis of the technology that will be used to accomplish the verification goal in order to ensure that diversion of a significant quantity of material can not be hidden in measurement error. Potential technologies for use in the verification regime are presented in chapter 5.

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[1] There are no international controls which require the application of safeguards on the export of tritium production technology.

4.1 Production Assumptions of Regime

In order to develop a safeguard regime, quantitative assumptions regarding production goals and other physical parameters must be made. Although agreed production limits fall under the purview of strategic analysis and national politics, assumptions must be made in order to proceed in the safeguards analysis. Thus, for the present analysis, two production assumptions will be considered.[2]

The first limit corresponds to sustaining half of the current nuclear weapons stockpile (approximately 12,500 weapons). This would require sustaining 40 kg of tritium. Since tritium has a half life of 12.32 years, it will have an annual decay rate of 1-exp(-.693/12.32) = 5.5 %. Thus, the annual production limit would be 2.2 kilograms per year.

The second limit corresponds to sustaining a nuclear weapons stockpile required for minimum deterrence. Although there is no consensus regarding the number of weapons required for this purpose, the present analysis assumes this to be near 1000 weapons[3]. In order to sustain 1000 tritium burning weapons, annual required tritium production would be

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[2] If tritium production is accepted as an arms control measure, and verification systems are put in place, production limits can be increased or decreased depending on existing diplomatic and strategic relations.

[3] Former Secretary of Defense McNamara, in Blundering into Disaster (p.123), suggests that the U.S. and U.S.S.R. would need less than 500 strategic warheads each for finite deterrence.

approximately 200 grams[4], assuming remaining weapons and associated tritium have been verifiably destroyed.[5] Using the rule of thumb presented in chapter three for reactor requirements for tritium production, it is clear that both of these quantities can be produced with the operation of one production reactor in each country.

4.2 Objective of Tritium Safeguards Regime

The objective of the tritium safeguards regime is to detect the excess production of a significant quantity of tritium with high confidence and in a timely manner, and to detect the diversion of a significant quantity of tritium from peaceful applications.[6]

What is a Significant Quantity?

A "significant quantity"(SQ) in a superpower tritium verification regime has fundamentally different implications

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[4] see chapter three for calculation of the amount of tritium used in nuclear weapons.

[5] Although arms reductions below these limits are possible, such reductions are not currently considered feasible, and tritium verification for such low limits would require strict limitations on peaceful uses of tritium.

[6] It is probable that tritium used in peaceful applications can be quickly re-purified. The key issue in the area of peaceful uses is the time required to assemble a significant quantity from dispersed sources. For example, the 62 grams of tritium that were supplied for domestic non-military use in 1986 were supplied to 16 different companies and research institutes in 130 shipments. These supplies were further divided in the supply of commercial goods such as self-illuminating signs.

than in an IAEA safeguards regime. In the IAEA regime, a SQ is based on physical parameters; it corresponds roughly to the amount of material required to produce one nuclear weapon. In the superpower tritium verification regime, the value of a SQ is dependent on existing strategic capabilities. A SQ would correspond to the amount of material required to make a strategically or politically significant difference in superpower arsenals. A SQ with nuclear stockpiles at 20,000 is much different than a SQ with nuclear stockpiles at 1000 weapons.

Moreover, the definition of SQ depends on the generic goal of the treaty as well as the assumptions of the regime. If a clandestine stockpile of tritium is assumed a priori, then a SQ would correspond to that amount of tritium required simply to <u>sustain</u> a significant number of weapons (i.e. 5.5% of significant number of weapons). Alternatively, if a clandestine stockpile is not assumed, a SQ would correspond to the amount of tritium required to <u>create and sustain</u> a clandestine stockpile. In addition, if the purpose of the tritium limitation is to constrain the development of advanced weapons requiring increased tritium production rates, the value of SQ may be larger. However, since the primary benefit of tritium limitations described in chapter one is the fact that its control can mitigate concerns about clandestine stockpiles of fissile materials, the SQ that is assumed in this analysis is that required to <u>sustain</u> a

clandestine stockpile.

In the case of 12,500 weapons, this analysis assumes that a SQ would correspond to sustaining 500 excess weapons. Thus an annual SQ would be approximately 100 grams. In the case of stockpiles required for minimum deterrence, a SQ is assumed to correspond to sustaining 200 excess weapons or approximately 40 gram of excess production per year.[7]

What is a Timely Manner?

Following IAEA practice, we define detection in a timely manner as detection before a significant quantity of tritium produced in violation of the regime can be deployed in a weapon. Since the definition of the significant quantity is dependent on strategic conditions, the definition of "timely manner" is also dependent on strategic conditions.

The time required for a nation to produce an excess significant quantity of tritium without detection depends on the verification regime. For example, in the absence of verification, a 2400 MWth reactor could produce 100 grams of

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[7] The SQ with 1000 weapons each corresponds to a greater fraction of stockpile than the SQ with 12,500 weapons each because, if the superpowers are willing to reduce their stockpiles to 1000 weapons, it is assumed that they would have also adopted a strategy of minimum deterrence, whereby only numerical advantage that might yield a counterforce capability are significant. However, if the superpowers are not willing to reduce weapons stockpiles to these levels, it is assumed that they would be pursuing a nuclear strategy similar to the one they currently pursue, whereby incremental increases in stockpile number are believed to have political advantages. There will probably be considerable disagreement as to the exact number of weapons required for a counterforce capability under the former assumption.

tritium in approximately three days.[8] In the verification regime, however, insertion of quantities of lithium sufficient to produce tritium in excess of agreed limits would suggest an intention to violate the agreement.

4.3 Developing a Safeguards Approach

In developing a safeguards approach, the IAEA hypothesizes diversion scenarios and develops procedures that result in observable anomalies if a diversion is attempted.[9] Based on this analysis of diversion potential, the Agency establishes material balance areas and key measurement points for measurement, as well as locations for the emplacement of Containment/Surveillance devices.

The Agency typically requires and makes use of the States System of Accounting and Control (SSAC) in establishing the material balance. It does not verify every operator measurement, but rather a statistically based subset of such measurements. The Agency pursues this strategy primarily to conserve limited resources. In the current safeguards application, however, only one military facility will be safeguarded in each country, and it is feasible to verify every operator measurement. Thus, the SSAC would not

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[8] Using the rule of thumb that .0125 grams of tritium are produced per MWDth of reactor operation, 2,400MW*n*.0125 g/MWD = 100g; n=3.33 days.

[9] H. Gruemm, "Safeguards Verification - Its Credibility and the Diversion Hypothesis," IAEA Bulletin 25, no.4 (Dec. 1983): p.27.

be used directly in the establishment of the material balance. Rather, it would be used to facilitate inspector measurements, and to aid in determining the source of disputed production calculations.

4.4 Diversion Analysis

Within a safeguards regime, there are two general strategies for an adversary wishing to produce and divert significant quantities of nuclear material. The first is to report falsified measurements of the material balance. That is, to understate inputs or overstate outputs and inventory differences. In a tritium verification regime, this would include, for example, such measures as the over statement of Li^6 in lithium waste, the overstatement of tritium losses in production[10], or the production and irradiation of non-certified targets. The second general diversion strategy is the diversion of material within the MUF. That is, the reporting of unfalsified measurement results, but diverting within measurement error. This is generally a problem in large bulk-handling facilities such as plutonium reprocessing plants.

The general countermeasure for the first diversion

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[10] A key consideration in accounting for tritium is the verification of process losses, especially if such losses can exceed a significant quantity. Otherwise, the credibility of the verification regime would be undermined. As shown in section 3.4, in a well run facility, tritium losses are on the order of 4% of tritium production, and most of this can be accounted for by known mechanisms.

strategy is to verify reported measurements. The countermeasure for the second strategy is to close the material balance more frequently, i.e. weekly instead of yearly, and to use high accuracy measurement technology.

The feasibility of the falsification strategy is analyzed in this chapter, and that of diverting within MUF in chapter 5. From the discussion in chapter 5, it is clear that measurement error does not pose a major threat to the regime.

Assuming the existence of a State System of Accounting and Control (SSAC) that monitors both Li6 and tritium in the production cycle, seven potential diversion scenarios can be identified for the production cycle described in chapter three:

1) understate number of target/ assemblies fabricated in fuel fabrication facility and placed in core for irradiation;

2) understate number of target assemblies placed in core--assume target assemblies fabricated at clandestine facility;

3) understate Li^6 content of target assemblies (either by increasing isotopic enrichment of lithium or increasing weight % of lithium in aluminum alloy);

4) overstate Li^6/tritium content of waste;

5) undeclared production of tritium in Li control rods;

6) understate tritium content in extraction tank, or at other points in the purification/enrichment process;

7) operating reactor at higher than declared power in order to produce extra tritium from Li^6 targets.

As mentioned earlier, a dual approach of monitoring both

tritium extraction and Li^6 target depletion, combined with surveillance measures is required to preclude or detect all of these excess production scenarios. Table 4-1 summarizes countermeasures to each scenario.

Diversion Route	Measures to Ensure Detection
understate number of target assemblies fabricated at declared facility and irradiated at reactor	inspector surveillance during fuel reload/ continuous optical camera surveillance during reactor operation/ item count and seal application at fabrication facility/ measurement of tritium in extraction tank.
understate number of target assemblies placed in core that have been fabricated at clandestine facility	inspector surveillance during fuel reload/ continuous camera surveillance during reactor operation/ measurement of tritium in extraction tanks.
understate Li^6 content of target prior to irradiation	measurement of Li^6 content prior to irradiation/ application of seals to ensure only measured target assemblies placed in core/ measurement of tritium in extraction tank.
overstate Li^6 content of irradiated target	measurement of Li^6 content after irradiation/ measurement of tritium in extraction tank.
overstate tritium content of waste	sample waste
undeclared production in tritium control rods	same as for other targets
understate quantity of extracted tritium	sampling from extraction process tank/ Li6 balance in pre and post irradiated targets

Table 4-1
Summary of Diversion Analysis

4.5 Material Balance Areas and Key Measurement Points

In the safeguards approach, material balance areas and key measurement points would be as shown in Figure 4-2. The general lithium and tritium verification activities are explained in sections 4.5.1 and 4.5.2, and specific activities to be pursued at each KMP are described in section 4.7.

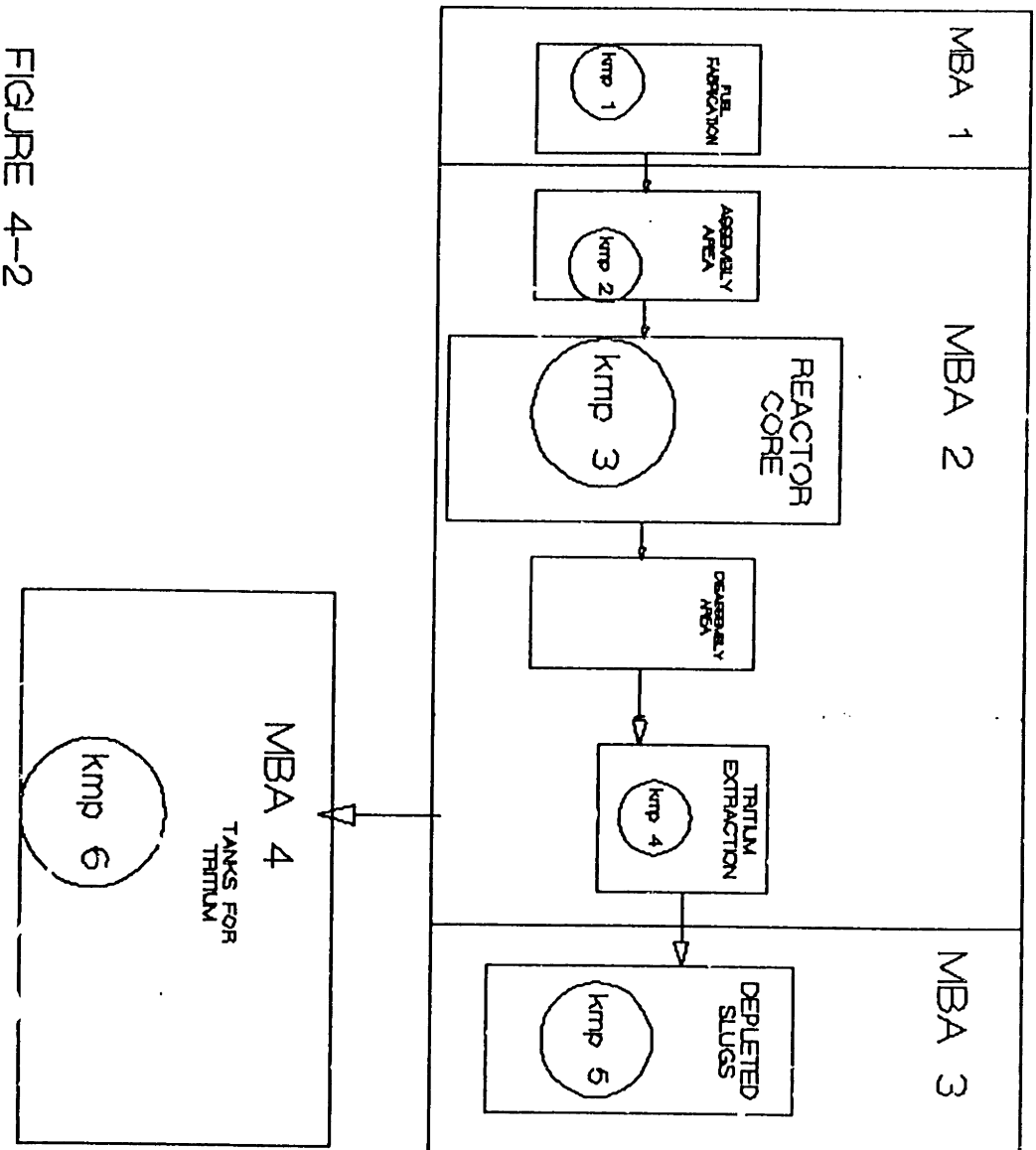
MBA 1
MBA 2
MBA 3
MBA 4
TANKS FOR TRITIUM
kmp 1
kmp 2
kmp 3
kmp 4
kmp 5
kmp 6
REACTOR CORE
TRITIUM EXTRACTION
DEPLETED SLUGS

FIGURE 4-2

4.5.1 Lithium Verification

Lithium verification is the key verification activity in the safeguards regime. By establishing a balance between Li^6 atoms prior to irradiation and after tritium extraction, the total amount of tritium produced can be determined, assuming that lithium targets are not covertly irradiated.

In accounting for Li^6, inspectors would determine the Li^6 content of the targets before and after irradiation. In each case four separate quantities must be measured or calculated:

-- weight fraction of lithium in target;
-- density of target;
-- enrichment of Li^6; and
-- volume of target.

4.5.2 Tritium Verification

Tritium measurements would be used to establish the tritium material balance. Moreover, they would complement lithium verification by aiding in the detection of unauthorized target irradiation, and providing a second check on the accuracy of lithium measurements. In accounting for tritium produced, the measurement point would be in the tanks in which tritium is accumulated directly after target heating. Inspectors would take gas samples for assay, and measurements to calculate the quantity of gas produced.

It should be noted that there are several potential key measurement points for tritium:

-- gas extraction point;
-- after purification;
-- after enrichment (flow from cascade); or
-- in container prior to shipment.

By verifying tritium content after it has been purified and enriched, the inspector would make excess production more difficult, as the diverter would then need to construct and operate a clandestine purification and enrichment facility for illicit targets irradiated in the safeguarded reactor. However, the verification approach must reflect the desire of treaty signatories to protect nuclear weapons stockpile information not covered by the treaty. Since all tritium in weapons requires periodic re-purification due to contamination of helium from tritium decay, allowing the monitoring of chemical purification and isotopic enrichment facilities could allow access to detailed information on stockpile requirements (i.e. management and tritium purity requirements).

Thus, monitoring beyond extraction process tanks might not be acceptable to either treaty party due to the potential for espionage.[11] This constraint, however, does not excessively detract from the safeguards approach, as no tritium is created after extraction.

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[11] Tritium might not be re-enriched and re-purified at the same facility as it is initially enriched and purified.

4.6 Containment/Surveillance

C/S supplements both the lithium and tritium measures. It includes both human and technical monitoring to ensure that only properly measured and approved targets are inserted in the reactor core and that tritium extracted from the targets is not diverted prior to measurement. Inspectors would be present in fuel fabrication facility to place tamper proof seals with serial numbers on fuel assemblies, and during reactor refueling operations to ensure that only approved targets are inserted and removed.

Surveillance devices such as optical cameras would be used to ensure that lithium targets are not inserted and/or removed from the reactor core when inspectors are not present. As discussed in chapter three, in order to insert or remove targets from the reactor core, the reactor must be shut down, and the upper plenum removed. Moreover, all components are charged and discharged using remotely operated cranes (see Figure 4-1). Both the charge and discharge machines have three masts to move equipment vertically in and out of the core. As such, optical surveillance devices monitoring the location of the upper plenum, as well as the charge and discharge cranes can ensure that lithium targets are not inserted when U.S. inspectors are not present.[12]

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[12] According to SRL employees, both the upper plenum and cranes could be adequately monitored from a process room viewing the closed core.

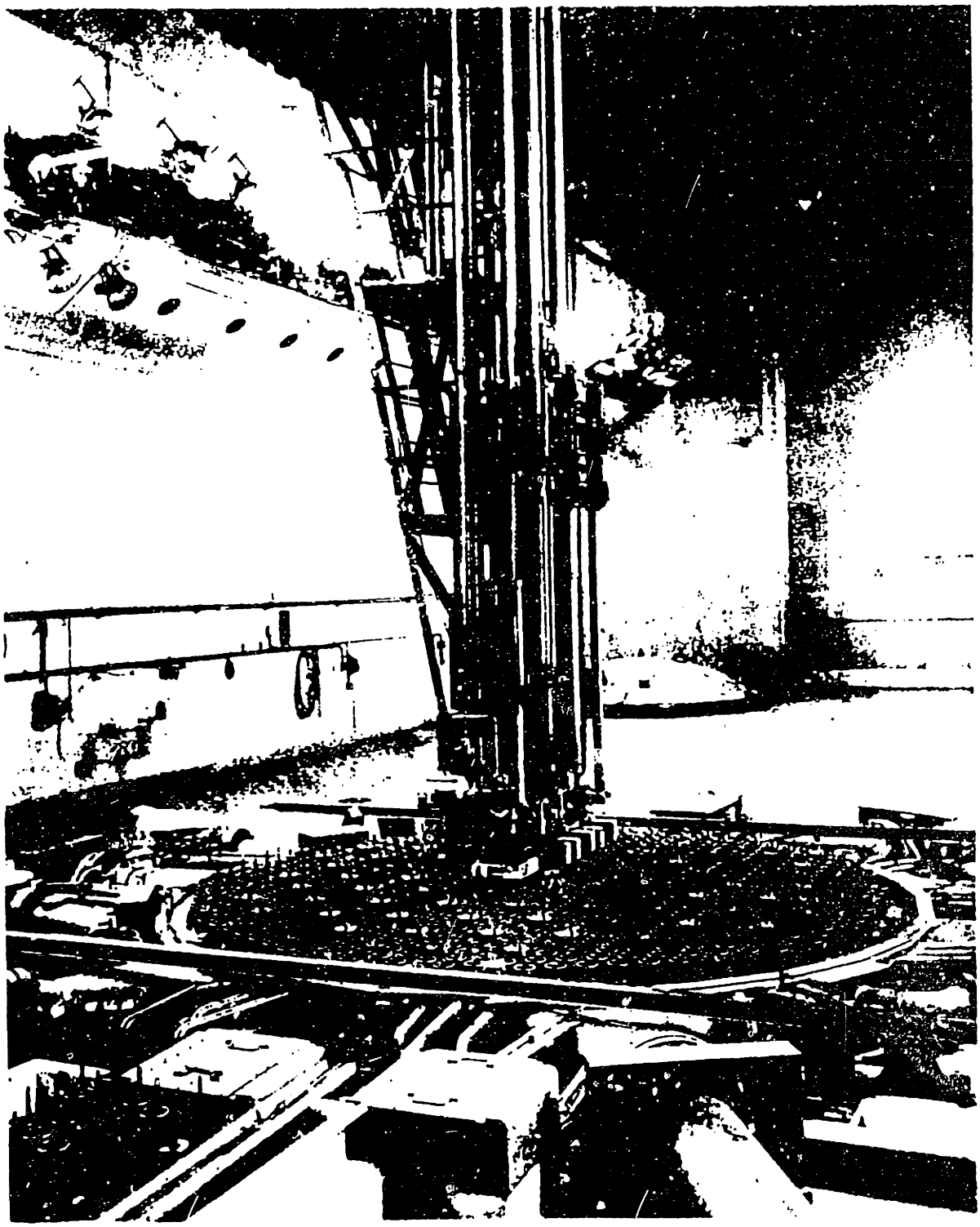

Figure 4-1
Charge and Discharge Machine Over Reactor

4.7 Specific Safeguards Activities

Specific safeguards activities in the above regime would include:

-- exchange of tritium production information;
-- initial inspection;
-- follow up meeting to agree on location of surveillance equipment and procedures for routine inspection;
-- continuous monitoring during fuel reload and extraction operations; and
-- routine inspections weekly to ensure proper operation of surveillance equipment and inspect fresh and spent fuel pond.

The initial verification activity would be the exchange of production information. This would include the identification of all facilities utilized for the production of tritium including:

-- facilities used for the fabrication of lithium targets;
-- nuclear reactors used for the production of tritium; and
-- facilities used for the extraction of tritium from irradiated targets.

This identification would include the location, purpose, description of the layout of each facility including locations of lithium and tritium, and a description of processes for accountancy and control of tritium and lithium. The information would further specify the chemical form, enrichment, and volume of all lithium target assemblies, as well as a technical justification for the quantity of lithium that would need to be inserted into the reactor core in order to produce the tritium limit.

For the tritium verification, the volume of the tank

used for accumulating extracted tritium, and a detailed description of procedures followed during fuel reload would be required. Moreover, a detailed description of the flows of tritium between the extraction oven and extraction tanks would be required.

An initial ad hoc inspection would be allowed to each treaty party to verify exchanged information, and determine specific location of key measurement points and location of surveillance equipment. In a later meeting, procedures would be established for application of seals and installation of surveillance cameras.

Routine inspections would be carried out weekly, based on the earlier calculation that shows that it would require a minimum of 3 days to produce an excess significant quantity of tritium, and an assumed time requirement of roughly 4 days to open the reactor, extract the targets, extract the gas from the targets, and purify, enrich, and deploy the tritium. Routine inspections would include item count, sampling and application of seals at KMP 1; item count at KMP 2 and KMP 4; and inspection of C/S equipment. Continuous inspector surveillance would be carried out during reactor reload, and during extraction activities. At these times, measurements would be taken at KMP 5, KMP 6, and an item count at KMP 3. The character of these measurements are discussed in chapter 6.

4.8 SSAC

As discussed in section 3.4, the IAEA, in applying safeguards, requires and makes use of the State's System of Accounting and Control (SSAC) of Nuclear Material. Although the SSAC would not be used to establish the material balance in the current application, it would be required to facilitate both the reconciliation of disputed measurements (i.e. so that the source of disputed tritium production quantities can be found and re-tested), and inspector measurements.

Specific information on the U.S. system of tritium accounting and control is not publicly available. Nevertheless, it is clear that the U.S. does have a SSAC for tritium, and it is likely that the U.S.S.R. does also. The Department of Energy requires that special nuclear material (SNM) have a measurement control program. Although tritium is not defined as SNM, due to its strategic and economic value, in 10 gram quantities, it is treated as SNM (category III). Moreover, SRL does have an accountancy program based on MBAs.[13]

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[13] Savannah River Laboratories uses a computer inventory control system that satisfies the reporting requirement of IAEA safeguards. The system contains both inventory files and system parameter files. AIMS maintains an up to date record of material by location (MBA). Inventory files are composed of records containing:

-- number of pieces;
-- commodity or gross weight;
-- ten material types.

The control system is based on Material Balance Areas

However, the U.S. does not have an adequate SSAC for lithium. Remote sampling and analytic equipment would have to be obtained by both parties. Thus, the U.S. (and possibly the Soviet Union) would have to expand their SSAC to include Li^6. French sampling methods could be used as a guide for both countries.

(called Control and Balance Account). Moreover, AIMS is designed to produce several types of reports on short notice which include beginning inventory, ending inventory, and transfer records. A material balance is calculated for all elements and isotopes on each transaction. Shipper and receiver weights must be the same.

CHAPTER FIVE

TRITIUM AND LITHIUM MEASUREMENT DEVICES AND CONTAINMENT/SURVEILLANCE EQUIPMENT

The previous chapter presented a safeguards verification regime that requires devices for measuring lithium and tritium, as well as containment/surveillance (C/S) equipment. Although C/S equipment is regularly used by the IAEA, tritium and lithium measurement devices have not been used by the IAEA, and have had little application outside of government tritium production facilities. The present chapter reviews tritium and lithium measurement devices and C/S equipment that could be used in the verification approach.[1]

5.1 Tritium Measurement Systems

Available information on the tritium measurement devices used at U.S. tritium production facilities and national laboratories has been reviewed; devices which could potentially be used in the verification approach are:

-- Calorimeters;
-- Mass Spectrometers;
-- Thermal Conductivity analyzers;
-- Ionization Chambers; and
-- Beta scintillation counters.

Each device offers distinct advantages and disadvantages that are summarized in Table 5-1. In general, in U.S. production

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[1] When available, the producer and model number of measurement devices used in U.S. laboratories are provided.

SYSTEM TYPE	CURRENT USE	PRIMARY ADVANTAGES	PRIMARY DISADVANTAGES
CALORIMETER	DETERMINE CONTENT OF PURIFIED-ENRICHED-PACKAGED GAS	MOST ACCURATE MEASUREMENT /ELIMINATES PROBLEM OF SAMPLE INHOMOGENEITY	TIME CONSUMING/ REQUIRES LARGE SAMPLES
P V T MASS SPECTROMETER	VARIOUS STAGES /EXTRACTED GAS /PURIFIED GAS /ENRICHED GAS	ACCURATE (±1%)/ USED AT SRL FOR ACCOUNTABILITY	SEVERAL MEASUREMENTS REQUIRED FOR ACCURACY
THERMAL CONDUCTIVITY	DURING ENRICHMENT	CONTINUOUS MEASUREMENT	PRIMARILY USEFUL WITH BINARY GAS
IONIZATION CHAMBER	AIR AND EXHAUST MONITOR/WASTE GAS	STURDY/SIMPLE/ACCURATE FOR LOW TRITIUM CONCENTRATIONS	NOT ACCURATE FOR HIGH TRITIUM CONCENTRATIONS
SCINTILLATION DETECTOR	?	ACCURATE FOR CONCENTRATIONS UP TO 80%	CURRENT USE UNCLEAR

TABLE 5-1 COMPARISON OF TRITIUM MEASUREMENT TECHNOLOGIES

facilities, low tritium concentrations are determined with ionization chambers, and high tritium concentrations are determined by mass spectrometry[2] and calorimetry[3].

Due to its accuracy, versatility, and widespread use, the Pressure, Volume, Temperature/Mass Spectrometer (PVT/MS) method of tritium measurement is probably the best choice to monitor tritium gas from the extraction facility. At Savannah River Laboratories, for example, MS is the primary method of direct control of the production process. Ionization chambers, however, are the most appropriate technology for monitoring low levels of tritium in the air and waste streams, if required.

5.1.1 PVT/MS Tritium Measurement

In the PVT/MS measurement technique, tritium content is determined from the equation:

$$g = \frac{mxpV}{zRT} \qquad (1)$$

where:
g=the number of grams of tritium;
x=mole fraction of tritium determined by MS;
p=the pressure of the gas (torr);
V=the volume of the gas (liter);
T=the temperature of the gas (K);
R=the universal gas constant=62.4 torr-L/gmole-K

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[2] See appendix A for a description of mass spectrometer design and operation.

[3] Calorimeters measure the power output of a quantity of gas. The mass of tritium in the gas is calculated by dividing the power output by the specific power of tritium (0.342 W/g).

m=molecular weight=6.032 g/gmole
z=compressibility factor=1 for low pressures and ambient conditions.[4]

The total number of atoms present are determined from measured gas pressure, volume and temperature, and the isotopic content is determined by mass spectrometry.[5] Using the accuracies quoted below in sections 5.1.1.1, 5.1.1.2, 5.1.1.3, and 5.1.1.4, for currently used systems, the overall accuracy of the technique is: $(0.64^2 + 0.52^2 + 0.20^2 + 0.30^2)^{.5} = \pm 0.90\%$ (2 sigma)

5.1.1.1 Mass Spectrometer

Tritium analysis is performed at SRL using Consolidated Electrodynamics Corporation Model 21-201 and Model 21-620 mass spectrometers and associated sampling systems.[6] Both a viscous sampling system and a molecular sampling system are used.[7] The relative accuracy of the mass spectrometers is ±0.64% (2 sigma)[8]. A sample can be analyzed

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[4] Charles Lindsay, Ronald Sprague, and Jeffrey Brandenburg, "A Measurement Control Study for Tritium Gas," Mound Laboratory report number MLM-3441, July 8, 1986, p.7.

[5] Lindsay, p. 7.

[6] A newer MS may currently be used to analyze gas samples.

[7] In a viscous sampling system, the quantity of gas flowing through the leak assembly is sufficient to ensure that collisions between molecules limit the rate of flow and so the rate of flow is dependent on the viscosity of the mixture. In a molecular system, the rate of flow of each gas is inversely proportional to the square root of its molecular weight.

[8] W.J Jacober, D.A. Orth, and G.W. Earle, "Tritium Control Technology-Separations Operation," Savannah River Laboratory report number DPSPU-73-30-7, May 1973, p. 11.

in five minutes in the magnetic scanning technique and in 2 minutes by voltage scanning.[9]

This system, or one very similar to it can be used to analyze the samples of gas obtained from the production process. Gas either would be removed from Soviet/U.S. process tanks and transferred to U.S./Soviet laboratories, or measurement could be accomplished in situ if each country allows the installation of a safeguarded laboratory in their production facility for dedicated use by the other party.

5.1.1.2 Pressure Measurements

Pressure measurements would be determined in situ using pressure transducers. At SRL, the pressure transducers used for tritium accountability are strain gage-type, manufactured by Statham, model number PA 824, with a measurement range of 0 to 40 psi.[10] At Mound laboratory, the strain gage pressure transducers are manufactured by Sensotec, and have a uncertainty of ±0.103 psi (2 sigma) in a calibration range up to 50.0 psi[11]. Since the design basis for most systems is about 50% of full scale[12], the relative accuracy for a typical pressure measurement at SRL, would be about ±0.52% (2 sigma).

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[9] E.L. Albenesius and L.H. Meyer, "Analytic Techniques for the Use and Control of Tritium at Savannah River," SRL report DP-771, September 1962, p. 15.

[10] Jacober, p.9.

[11] Lindsay, p. 15.

[12] Lindsay, p. 24.

5.1.1.3 Temperature Measurements

Very accurate temperature measurements at SRL are accomplished using special Brown resistance thermometer bulbs with a temperature range of 18 C to 30 C. The devices are accurate to ±0.1 C at 25 C, yielding a relative accuracy of ±0.03%. At Mound laboratory, temperatures are taken for tritium accountancy using either a thermocouple or platinum resistance thermometer. A typical accuracy of a Mound thermometer is about ±0.2% at 25 C.[13]

5.1.1.4 Volume Measurement

The volume of the tank for tritium, presumed constant, would be determined during the initial ad hoc inspection using pressure and temperature measurements[14]. At Mound facility, the uncertainty of volume measurements ranges from about ±0.1% on tanks less than 10 liters, to about ±0.3% on larger tanks of about 200 liters.[15] The larger tank size uncertainty is used in the calculation is 5.1.1, since production processes at SRL probably utilize large containers.

5.1.2 Ionization Chambers

Ionization chambers can be used in the measurement of tritium content in air and waste gas streams that range from

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[13] Lindsay, p. 24.

[14] It may prove advantageous to require the use of tritium tanks supplied by the inspecting country or standardized tanks agreed on by both countries.

[15] Lindsay, p.24.

10^{-4} to 1 mol %. Ionization chambers are simple, sensitive and sturdy devices filled with gas. They can be adapted to either flow or static systems. The ion chamber shown in Figure 5-1, is used at SRL and is constructed of glass with silvered inner surfaces as electrodes. The tritium decay causes a current in the detector that is amplified and measured with a Beckman microameter. The lower limit of tritium measurement is determined by background current caused by tritium sorbed to the walls of the chamber. The upper detection limit is set to overlap with the lower detection range of the mass spectrometer (.01 mol %).[16]

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[16] Albenesius, p. 10.

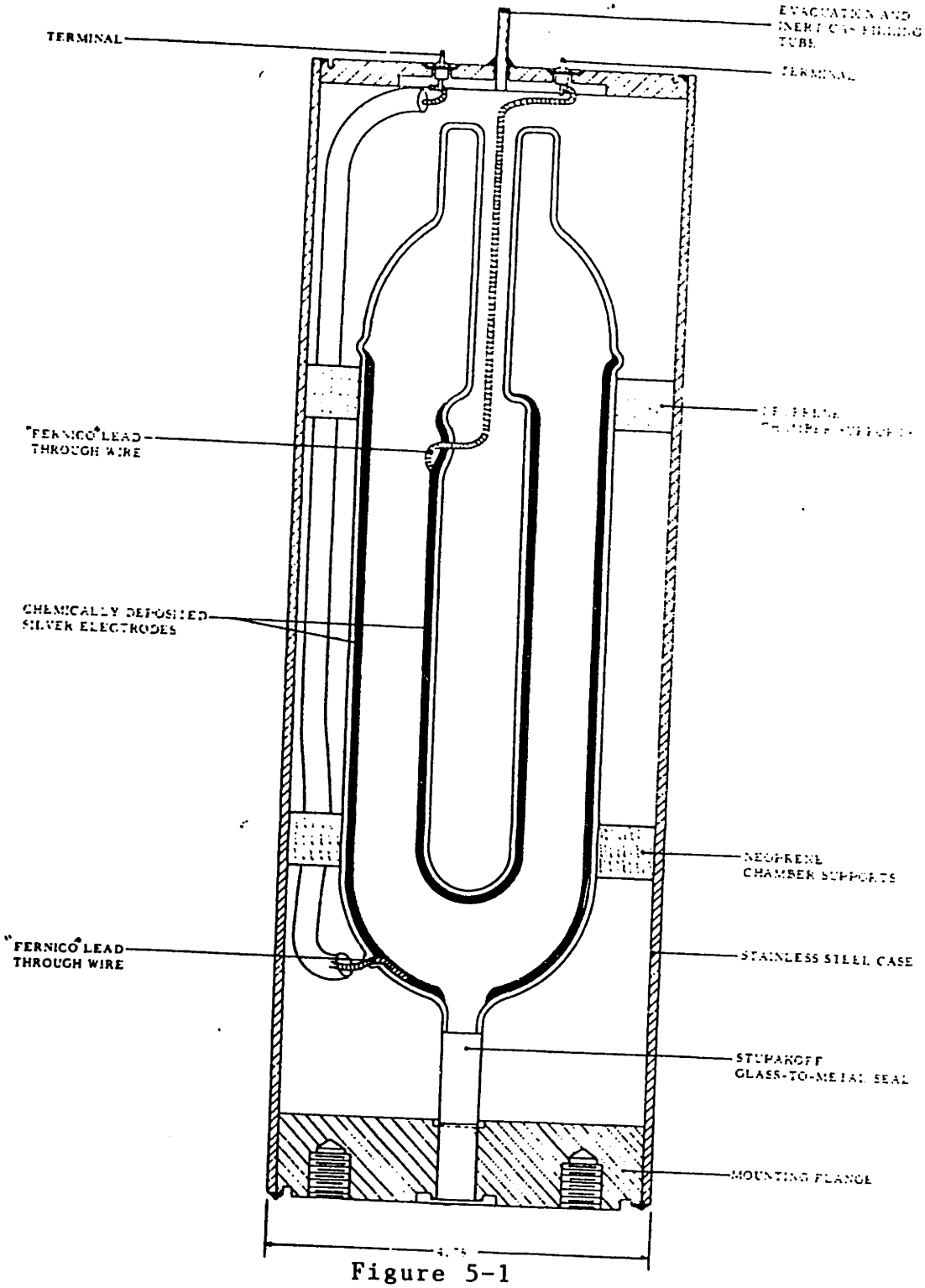

Figure 5-1
IONIZATION CHAMBER FOR PROCESS CONTROL

5.2 Lithium Measurement Systems[17]

Lithium measurement techniques are not as well developed as tritium techniques because lithium accountancy is not priority in U.S. laboratories. Thus, the costs and limitations of each technique are not as clear as those for the tritium measurement techniques. Several techniques for lithium measurement have been tested:

-- neutron activation;
-- mass spectrometry;
-- nuclear magnetic resonance spectroscopy;
-- optical spectroscopy; and
-- neutron transmission.

In all these technologies except neutron transmission, (milligram) samples would be take from enriched and depleted lithium targets, transferred to the investigator's laboratory, and converted to an appropriate chemical form for

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[17] The verification approach requires devices for determining the isotopic content of lithium targets, as well as weighing lithium targets and determining their chemical composition (i.e. lithium content). Since mass scales are widely used by the IAEA, and have errors that are more than an order of magnitude less than those in devices for determining lithium isotopic content, mass scales are not addressed in detail in this study. For example, according to Aurbach (see note # 30), a precision balance can increase the precision of mass measurement from 0.1% for standard scales to better than 0.01%. The method of determining the chemical composition of the lithium target will depend on the composition of the target material. If the target is lithium-aluminum alloy, the lithium fraction can be determined from the density of the material or by some other form of chemical analysis. This measurement would also have an error that is much smaller than the error in measurement of lithium isotopic content. If the target material is a chemical compound, such as lithium aluminate, then the lithium fraction is known from the chemical equation of the compound.

analysis.

For the purposes of this verification regime mass spectrometry would probably be the best technique. This technique is accurate, and its limitations appear less problematic than those of the other techniques. However, each form of assay offers advantages that are described below, and their application should be further explored. The relative advantages and disadvantages of lithium measurement devices are summarized in Table 5-2.

SYSTEM TYPE	PRIMARY ADVANTAGES	PRIMARY DISADVANTAGES
MASS SPECTROMETRY	ACCURATE (±1%)/ WIDELY USED/	POSSIBLY COMPLICATED SAMPLE PREPARATION
NMR	SAMPLE PREPARATION LESS COMPLICATED THAN MS/ POTENTIALLY ACCURATE TO ±1%	UNPROVEN APPLICATION
ACTIVATION	ACCURATE (< ±1%)/ POTENTIALLY LESS EXPENSIVE THAN MS	REQUIRES ACCELERATOR: CAN NOT HAVE IN SITU MEASUREMENT
NEUTRON TRANSMISSION	IN SITU MEASUREMENT/ ALREADY USED AT SRL/ ACCURATE	MEASUREMENT OF IRRADIATED TARGETS MAY NOT BE FEASIBLE

TABLE 5-2 COMPARISON OF LITHIUM MEASUREMENT TECHNOLOGIES

5.2.1 Mass Spectrometry

Mass spectrometers are widely used and are a very accurate form of lithium assay[18]. Although no information on mass spectrometers for lithium analysis in U.S. facilities is available, Scientists at the Bhabha Atomic Research Center (BARC), in India, have developed and tested a thermionic mass spectrometer for lithium enrichment determination. This device is appropriate for the verification system. At BARC, samples were composed of $LiNO_3$. In a 2 hour time period, a 10 micro-gram sample of lithium was analyzed with an accuracy of about ±1% (2 sigma).[19]

The device uses a single focussing thermionic source with multiple filament assembly. The mass spectrum is obtained by varying a magnetic field; the ion current is measured by a faraday cup with vibrating reed spectrometer or secondary electron multiplier coupled to a d.c. amplifier. Output is fed to a strip chart recorder and a TDC-312 computer.[20]

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[18] S.A. Chitamber et al, "Mass Spectrometric Analysis of Lithium," Bhabha Atomic Research Centre, Bombay, India, BARC report #976, 1978, p. 1.

[19] Chitamber, pp. 4-8.

[20]

Ion accelerating voltage	3KV
Source slit width	0.1 mm
Collector slit width	0.3 mm
Radius of curvature	21.4 cm
Angle of deflection	90^{o}
Resolution	425 measured at mass 235 and 238
Abundance sensitivity	5x10^5

Although mass spectrometry provides precise results, it can require complicated chemical sample preparation procedures, it is expensive, and is time consuming[21]. Most samples must be converted to a suitable inorganic salt such as $LiNO_3$ prior to measurement. Thus, other methods should also be investigated.

5.2.2 Neutron Transmission

Although neutron transmission is not cited as a method of lithium measurement in the current literature, prior to 1955 an instrument was developed by E.I. du Pont de Nemours & Co. to measure non-destructively the lithium content of entire natural lithium-aluminum-alloy target slugs for the Savannah River reactors. The device was designed to analyze slugs consisting of cylindrical pieces of the alloy 0.8 inches in diameter and 12 inches in length, with a natural lithium content of between three and seven percent. Good correlation was obtained between the neutron transmission measurement and the lithium content determined through chemical analysis.[22]

Although the fact that the device has been used at SRL lends credibility to its application, the device was probably used only to analyze the lithium targets prior to

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[21] J. Asher and M.T. Swinhoe, "An Activation Method to Determine the Isotopic Ratio $^6Li/^7Li$ in Lithium Compounds", Nuclear Instruments and Methods 213 (1983): p. 503.

[22] A.H. Dexter, "Measurement of Lithium in Target Slugs by Neutron Transmission," Savannah River Laboratory report DP-106, Feb. 1955, p.7.

irradiation. After target irradiation and tritium extraction, the target would be so deformed that measurement of target thickness, which is essential for neutron transmission measurements, would be difficult and inexact. Although target deformation prior to extraction is probably small, it is not clear if lithium measurement prior to extraction is feasible due to potential tritium losses.

Nevertheless, use of neutron transmission measurement would allow the assay of fresh targets at the reactor facility, and would thus reduce concern of target substitution between the target fabrication facility and the reactor. It is possible that problem of target deformation can be overcome by accumulating the spent melt in a mold of known dimensions. If this can be accomplished, neutron transmission may be the best lithium measurement technique for the verification regime.

5.2.3 Nuclear Magnetic Resonance[23]

Researchers at Chalk River Nuclear Laboratories, Canada, are investigating the application of nuclear magnetic resonance to the measurement of lithium enrichment.[24] The

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[23] NMR refers to phenomenon of resonance absorption of radio frequency electromagnetic radiation produced by simultaneously applying a magnetic field and radio frequency electromagnetic radiation to a sample of atoms. It is useful in lithium measurement because Li6 and Li7 atoms have different resonance absorption frequencies.

[24] Kenneth J. Franklin, James Halliday, Lynne Plant, and Allen Symons, "Measurement of the $^{6}Li/^{7}Li$ Isotope Ratio for Lithium Salts by FT NMR Spectroscopy," Journal of Magnetic Resonance, 67 (1986): p. 162.

researchers, using a coaxial NMR tube system, completed initial work with a sample of LiCl in water and pyridine in 1985. An estimate of the precision of enrichment level is approximately ±3% (2 sigma). The ultimate goal of the project is to carry out enrichment measurement in less than two hours with a precision of ±1% (2 sigma).[25] Nevertheless, the technology remains unproven and should not be relied on until its use becomes accepted by scientists in both the U.S. and U.S.S.R.

5.2.4 Activation

Two activation methods for determining the isotopic composition of lithium exist. One has been used at the Y-12 plant in Oak Ridge Tennessee to measure the isotopic ratios of Li^6 to Li^7 in lithium hydride by observing the ratio of energetic He^4 and He^5 atoms produced by the bombardment of the lithium by .8 MeV deuterons.[26] This method has an accuracy of better than ±1%. The second activation method[27] relies on the production of Be^7 by bombardment of the lithium sample with .8 MeV deuterons.[28] Li^6 density is calculated from the gamma radiation emitted during Be^7 decay. This method has an accuracy of roughly ±2-3%. The primary

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[25] Franklin, p. 165.

[26] The method depends on the reactions Li6(d,alpha)He4, and Li7(d,alpha)He5.

[27] J.L. Cochran, and J.T. Hill, "The Measurement of Lithium Hydride Enrichment," IEEE (1981), p. 1855.

[28] Li6 + d -->n + Be7

drawback of both of these methods is that a Van de Graff accelerator would be required to produce the energetic deuterons. It would not be feasible to locate such an instrument in situ.

5.3 C/S EQUIPMENT

As mentioned at the beginning of this chapter, the IAEA regularly uses C/S equipment. The two types of C/S devices that are required in a tritium verification regime are seals and film or TV cameras.

The principal seal used by the IAEA is a Type-E and Type X metallic seal. Although these seals are inexpensive and easy to apply, their integrity can not be checked in situ. New seals being investigated by the Agency include fiber optic seals, electronic seals, and ultrasonically verified seals that can be checked in situ.[29] Moreover, for identification purposes, the Agency is evaluating a BWR fuel assembly identification device that is designed to uniquely identify fuel assemblies during the entire fuel cycle.[30]

For optical surveillance, the Agency uses both film cameras and closed circuit television. The camera system most commonly used consists of two Minolta XL 401 Super 8 mm

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[29] A. von Baeckmann, "The Application of Modern Methods and Techniques in Safeguards Operations," IAEA Bulletin 23, no. 1 (March 1981): p.18.

[30] Clemens Auerbach, "Safeguards Instrumentation A Computer-Based Catalog, Second Edition," Brookhaven National Laboratory report number BNL-51450, April 1985.

film cameras with a tamper-resistant sealable enclosure and mounting bracket. These systems are self-contained and do not require power from the inspected facility. Closed circuit television (CCTV) is used when continuous surveillance is required, where radiation levels are high enough to damage film, or where recordings must be reviewed in situ.[31] CCTV is becoming more frequently applied by the Agency due to reliability problems with the Minolta film cameras.

A technique for remote verification of C/S equipment has been analyzed by the Agency for several years, but has not been adopted. The technique called recover (REmote COntinual VERification) consists of monitors connected to the C/S devices. Encrypted data from the monitors are transmitted continuously through telephone lines.[32] Such a system should be considered for superpower arms control verification. However the negotiation of provisions for the location of "black box" remote sensors on Soviet and U.S. territories could prove problematic, as it has in negotiating for the emplacement of remote sensors in the context of the Comprehensive Nuclear Test Ban Treaty (see chapter 1).

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[31] Aurbach, pp. 87-89.

[32] Baeckmann, p. 19.

CHAPTER SIX

SAFEGUARDS AT PEACEFUL FACILITIES

The second objective of a tritium control regime is to verify that tritium is not produced at declared peaceful facilities: power reactors or research reactors and via neutrons produced using accelerators. The latter issue is briefly discussed in chapter seven. Here we consider the feasibility of verifying the non-production of tritium at reactors. Although the U.S. tritium-producing reactors described in chapter three are specifically designed for this purpose, i.e., they are moderated by heavy water, operated at low temperatures, and do not produce electrical power, large quantities of tritium can be produced in any nuclear power reactor or high flux research reactor by altering core design, fuel design, fuel enrichment, and/or moderator. Moreover, smaller quantities can be produced without any changes in these characteristics.

The purpose of the current chapter is to illustrate methods of clandestine tritium production at power and research reactors; quantify production potential at such reactors; and finally to suggest general inspection techniques that would reduce the threat of such illicit production.

An accurate analysis of the tritium production potential in any given power reactor can only be obtained by a careful engineering study of the particular reactor system. However,

an analysis of each reactor is beyond the scope of this study. Since pressurized light-water (PWRs) are widely used in both the U.S. and U.S.S.R., this analysis concentrates on a generic PWR. Estimates of tritium production in boiling water reactors (BWRs) are also provided. Relevant differences between these reactors and other types of reactors operated by the U.S. and U.S.S.R. are discussed in section 6.6.

6.1 U.S. and Soviet Power Reactors

U.S. utilities and the Soviet government operate a large number of nuclear power reactors that could be used to produce tritium. A complete list of U.S. and Soviet power producing reactors appears in Appendix B.

6.1.1 U.S. Power Reactors

In the U.S., commercial reactors are primarily of two types: pressurized water reactors (PWRs) and boiling water reactors (BWRs). U.S. utilities currently operate 101 such light water reactors (LWRs) with an average power producing capability of 900 MWe. Of these, 67 are PWRs.[1] In addition to the LWRs, Public Service Company of Colorado operates one high-temperature gas-cooled reactor, and from 1962 until 1987 the United States Department of Energy operated the light-water cooled, graphite-moderated Hanford reactor for the

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[1] This number does not include the Seabrook or Shoreham reactors, both of which are complete, but are not currently operatin

production of plutonium as well as power.[2] Moreover, 17 new reactors are expected to be placed in commercial operation in the next ten years, 13 of which will be PWRs.[3]

6.1.2 Soviet Power Reactors

Soviet reactors are primarily PWRs or light water cooled, graphite moderated reactors (LGRs). Of the Soviet Union's 52 operating power reactors (average power of 697 MWe) 22 are PWRs and 23 are LGRs. In addition, two liquid metal fast breeder reactors (LMFBRs) and one small BWR are in operation. The Soviet Union also plans to place 21 new PWRs on line in the late 1980's and early 1990's, including two that were previously planned to be LGRs (Kostroma 3 and 4). According to a recent article in the magazine Nuclear Engineering International, "in the foreseeable future, PWRs will predominate (in the U.S.S.R.)." [4]

6.2 Brief Description of PWRs

Typically, PWRs are fueled with low enriched uranium (2-4%), operated at high temperature and pressure, and refueled off-load at yearly intervals during which 1/3 of the core is

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[2] During the mid 1960's, the Hanford reactor was shown to be capable of producing both tritium and plutonium in a coproduction mode. In the demonstration project, tritium was produced from lithium aluminate ceramic target material.

[3] "World List of Nuclear Power Plants," Nuclear News, February 1988, pp. 77-82.

[4] "Soviets Concentrate on the PWR and Work at Advanced Concepts," Nuclear Engineer International, June 1988, p. 33.

replaced. Lithium targets could be inserted in the core only during the refueling period. The following description of a typical 1000MWe PWR summarizes those design characteristics of PWRs that are necessary to understand the discussion of tritium production in section 6.3.

The core of a PWR has fuel assemblies of three different enrichments, e.g., 2.25%, 2.79%, and 3.29%. Each assembly is composed of 204 fuel rods and 21 guide tubes (thimbles) in a 15x15 array. The central guide tube contains nuclear instrumentation while the remaining 20 tubes could:

-- remain empty for control rod insertion;
-- contain burnable poison rods[5]; or
-- remain unused and therefore blocked by a thimble plug.

About one third of the assemblies in the core contain control rod assemblies.[6] A typical PWR reactor vessel and fuel assembly are shown in Figures 6-1 and 6-2. As will be discussed in section 6.3, the control rod guide tubes, as well as fuel tubes provide convenient locations for the insertion of lithium targets. Locations outside of the core, but within the pressure vessel, e.g., in positions around the core barrel, could also be used to produce smaller quantities

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[5] Burnable poison rods are utilized in LWRs to help control long-term changes in core reactivity due to fuel burnup and thus facilitates more uniform burnup of fuel. They are typically composed of borosilicate glass or boron carbide.

[6] Samual Glasstone and Alexander Sesonske, Nuclear Reactor Engineering (New York: Van Nostrand Reinhold Company, 1981), p. 742.

of tritium.[7]

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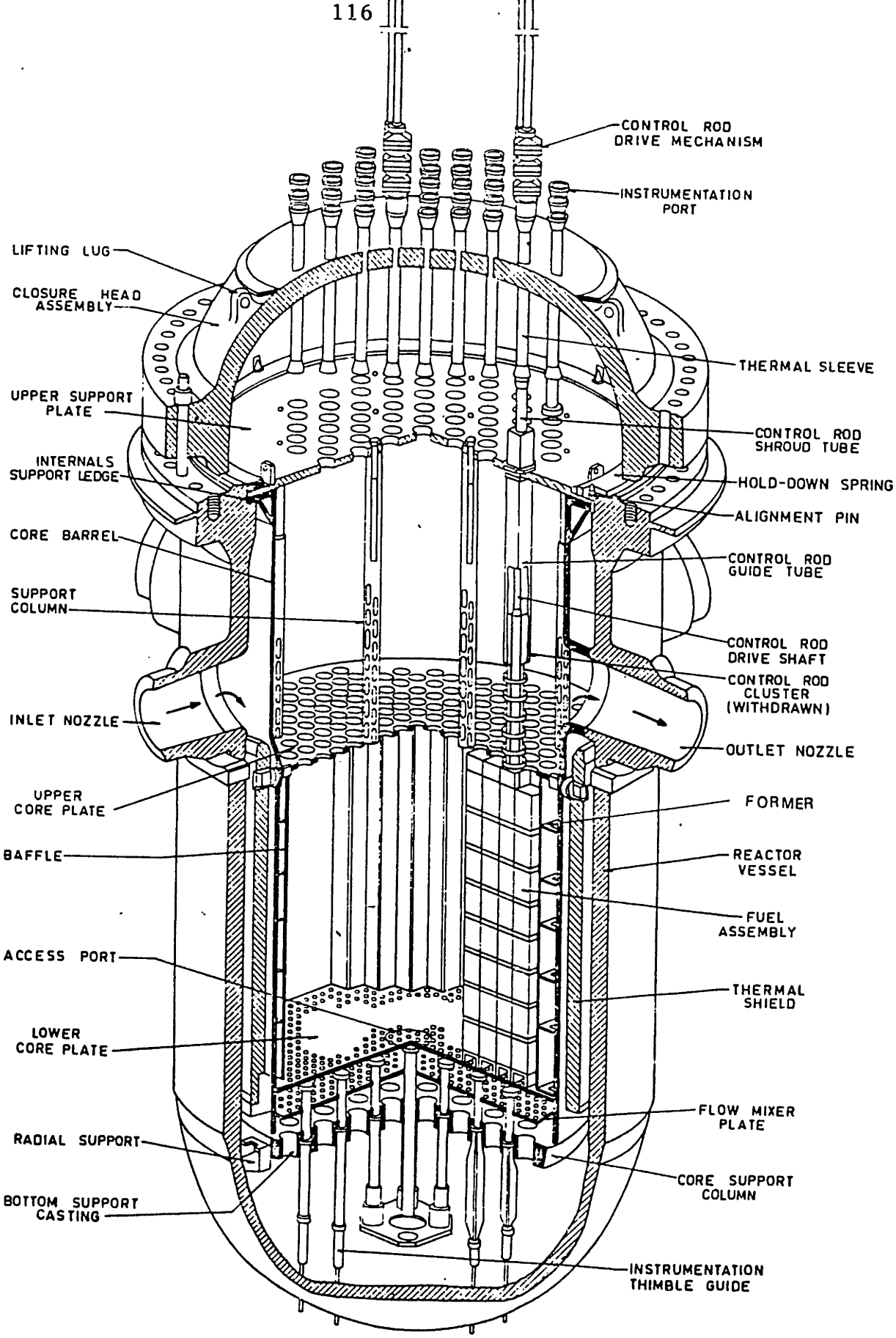
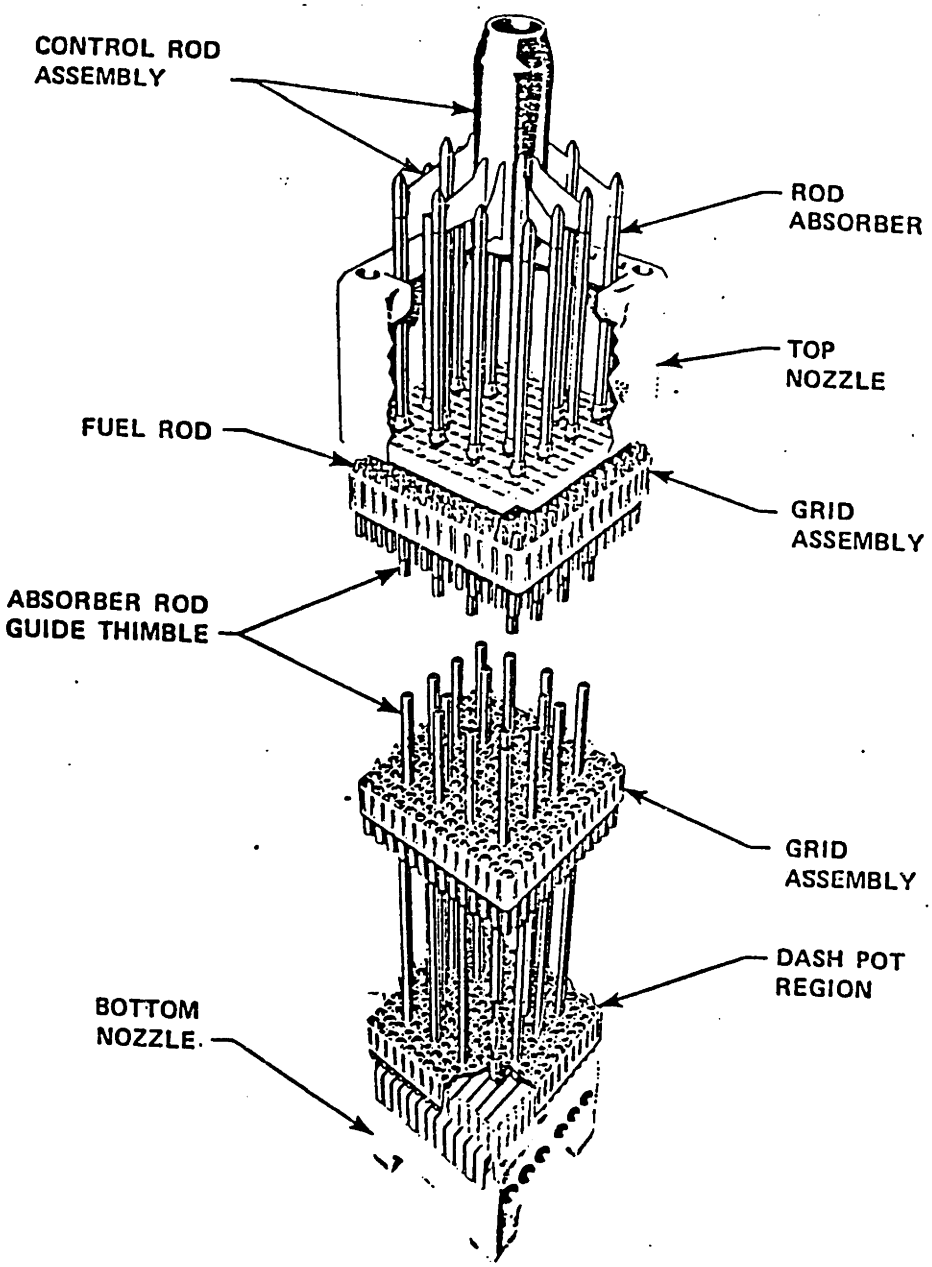
[7] M.-S. Lu, R.-B. Zhu, M. Todosow, "Unreported Plutonium Production in Light Water Reactors," Brookhaven National Laboratory report number ISPO-282, TSO-88-1, February 1988, pp. 2-1,2-2.



Reference PWR reactor vessel internals.

Source: Lu et al, p. 2-26

Figure 6-1



Typical PWR fuel assembly with rod cluster control assembly (14 x 14 assembly).

Figure 6-2

Source: Lu et al, p. 2-13.

During refueling fresh-fuel assemblies are transferred from the fuel pool to the containment fuel-handling pool via an underwater transfer tube. New fuel assemblies with burnable poisons are loaded directly into the reactor core. However, fuel assemblies that receive thimble plug or control rod assemblies are first placed in the control-rod-cluster fixture inside the containment fuel-handling pool where such assemblies are installed. A thimble-plug assembly or control rod assembly can also be transferred between fuel assemblies in the core. Burnable poisons are used for only one cycle. Used burnable poisons are transferred to a fuel assembly that is to be discharged.[8]

6.3 Tritium Production in PWRs

The three characteristics most relevant to the potential for tritium production in power reactors are:

> -- the power level of reactor operation (as indicative of the neutron flux in and around the core);
> -- the amount of space in and around the core in which targets could be inserted; and
> -- the excess reactivity of the core (i.e. the amount of reactivity in control rods, burnable poisons, and in neutron absorbing additives in the moderator and coolant);

Of these, only the last can be substantially altered without fundamental design changes in the reactor core and cooling capabilities. Thus, the tritium production capability of a

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[8] Lu, p. 2-7.

given reactor can be considered constant.

Tritium could be produced in a PWR by various means:

-- use of fuel assemblies specially designed for tritium production;
-- use of conventional fuel assemblies that have had fuel replaced by lithium targets;
-- insertion of lithium targets in guide tubes in place of thimble plugs or burnable poisons; and
-- insertion of lithium targets in empty regions outside the reactor core.

These methods of production are discussed in sections 6.3.2, 6.3.3, and 6.3.4 and 6.3.5, respectively.[9]

In discussing the potential for tritium production in power reactors, one must be aware of the relevant initial assumptions or "ground rules". The production potential of a given reactor will depend on the extent to which tritium production will be allowed to affect the normal functioning of the reactor. At one extreme, a power reactor could be altered to such an extent (i.e. change in fuel assembly design, fuel enrichment, operating cycle length, or even moderator) that it more closely resembles a production reactor than a power reactor. Of course, drastic changes could be easily detected. At the other extreme, the ground rules might stipulate that tritium production should not change reactor operating parameters above the normal uncertainty in operator measurements. The former ground rules most closely correspond to the discussion in section

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[9] However, specific reactors may have additional locations for lithium target insertion. For example, the Shippingport PWR has empty grid spaces around the core, that comprise convenient locations for lithium irradiation.

6.3.2, while the latter ground rules closely correspond to the discussion is section 6.3.4.

6.3.1 Lithium Target Composition

The target material that would be used for producing tritium in a LWR would probably not be the same material that is used in dedicated tritium production reactors. The higher operating temperature and rigid operating schedules of existing PWRs affect target requirements. For example, the lithium target material developed for the production of tritium in the Hanford power producing reactor was lithium aluminate. This material was chosen over the lithium aluminum alloy used in the SRL reactors, even though fabrication and extraction techniques for the alloy had already been developed, because safety considerations required a target that would not melt in the event of an accident that might raise the temperature of the target to 1100 C. The alloy used in the SRL reactors melts at roughly 635 C whereas lithium aluminate (LiAlO2) has a melting point above 1900 C.[10]

In addition, as a tritium production reactor operates, gas pressure from the generated tritium builds up within the capsule containing the lithium target. The target must be

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[10] Albert Kishbaugh, "Extraction of Tritium from Lithium Aluminate Targets", Savannah River Plant report # DP-1058, August 1966, p.1.

removed before the pressure becomes so great that the capsule fails. The required frequency of target removal would interfere with the normal operation of a LWR.[11]

These considerations have led Cawley et al to patent a lithium aluminate/zirconium material for the production of tritium in power reactors.[12] The material consists of up to 10 volume percent lithium aluminate particles (between 100 and 500 micrometer in diameter) imbedded in a zirconium matrix (Figure 6-3). The advantage of this material is that tritium produced from the lithium is absorbed by the zirconium, thereby reducing the gas pressure within the capsule containing the material.

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[11] William Cawley and Turner Trapp, "Lithium Aluminate/Zirconium Material Useful in the Production of Tritium," U.S. patent 4,475,948, Oct 9,1984.

[12] Cawley, U.S. patent 4,475,948.

United States Patent [19]

Cawley et al.

[11] **Patent Number: 4,526,741**

[45] **Date of Patent: Jul. 2, 1985**

[54] **FUEL ASSEMBLY FOR THE PRODUCTION OF TRITIUM IN LIGHT WATER REACTORS**

[75] Inventors: **William E. Cawley; Turner J. Trapp,** both of Richland, Wash.

[73] Assignee: **The United States of America as represented by the United States Department of Energy,** Washington, D.C.

[21] Appl. No.: **503,129**

[22] Filed: **Jun. 10, 1983**

[51] Int. Cl.³ G21C 3/00

[52] U.S. Cl. 376/185; 376/209; 376/435; 376/447; 376/449; 376/455

[58] Field of Search 376/185, 435, 447, 431, 376/189, 455, 209, 419, 449

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"A Brief Summary of Different Tritium Sources for a Tritiumless Hybrid Reactor", Moses, p. 7, Oct. 1979.

Primary Examiner—Deborah L. Kyle
Assistant Examiner—Dan Wasil
Attorney, Agent, or Firm—Robert Southworth, III; Judson R. Hightower

[57] **ABSTRACT**

A nuclear fuel assembly is described for producing tritium in a light water moderated reactor. The assembly consists of two intermeshing arrays of subassemblies. The first subassemblies comprise concentric annular elements of an outer containment tube, an annular target element, an annular fuel element, and an inner neutron spectrums shifting rod. The second subassemblies comprise an outer containment tube and an inner rod of either fuel, target, or neutron spectrum shifting neutral.

7 Claims, 1 Drawing Figure

United States Patent [19]

Cawley et al.

[11] **Patent Number: 4,475,948**

[45] **Date of Patent: Oct. 9, 1984**

[54] **LITHIUM ALUMINATE/ZIRCONIUM MATERIAL USEFUL IN THE PRODUCTION OF TRITIUM**

[75] Inventors: **William E. Cawley; Turner J. Trapp,** both of Richland, Wash.

[73] Assignee: **The United States of America as represented by the Department of Energy,** Washington, D.C.

[21] Appl. No.: **488,825**

[22] Filed: **Apr. 26, 1983**

[51] Int. Cl.³ F16N 57/04

[52] U.S. Cl. 75/230; 376/146; 376/202; 420/422

[58] Field of Search 75/230; 376/146, 202; 420/422; 501/105

[56] **References Cited**

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3,100,184	8/1963	Abraham	376/146

Primary Examiner—Leland A. Sebastian
Attorney, Agent, or Firm—Robert Southworth, III; Richard E. Constant; Michael F. Esposito

[57] **ABSTRACT**

A composition is described useful in the production of tritium in a nuclear reactor. Lithium aluminate particles are dispersed in a matrix of zirconium. Tritium produced by the reactor of neutrons with the lithium are absorbed by the zirconium, thereby decreasing gas pressure within capsules carrying the material.

6 Claims, No Drawings

Figure 6-3

6.3.2 Altered Assembly Design

Tritium could be produced in LWRs in fuel/target assemblies that resemble those of production reactors. Employees of the U.S. Department of Energy already have patented a fuel assembly for the production of tritium in light water reactors that would use the lithium aluminate/zirconium target material discussed in section 6.3.1 (Figure 6-3).[13]

The fuel/target assembly (Figure 6-4) consists of two intermeshing arrays of subassemblies. The large subassemblies comprise concentric annular channels for cooling water, lithium targets, fuel, and a zirconium rod. The second, smaller subassemblies consist of an outer channel for cooling water, and an inner channel for a rod of either fuel, target material, or zirconium.

Tritium production based on this design would be more efficient than production in existing PWR fuel assemblies (see section 6.3.3). In the latter case, neutrons formed in the fuel region have a greater probability of being absorbed before reaching the lithium target. However, such a radically altered fuel assembly design could easily be detected by visual inspection. More credible production scenarios, with more demanding ground rules are discussed below.

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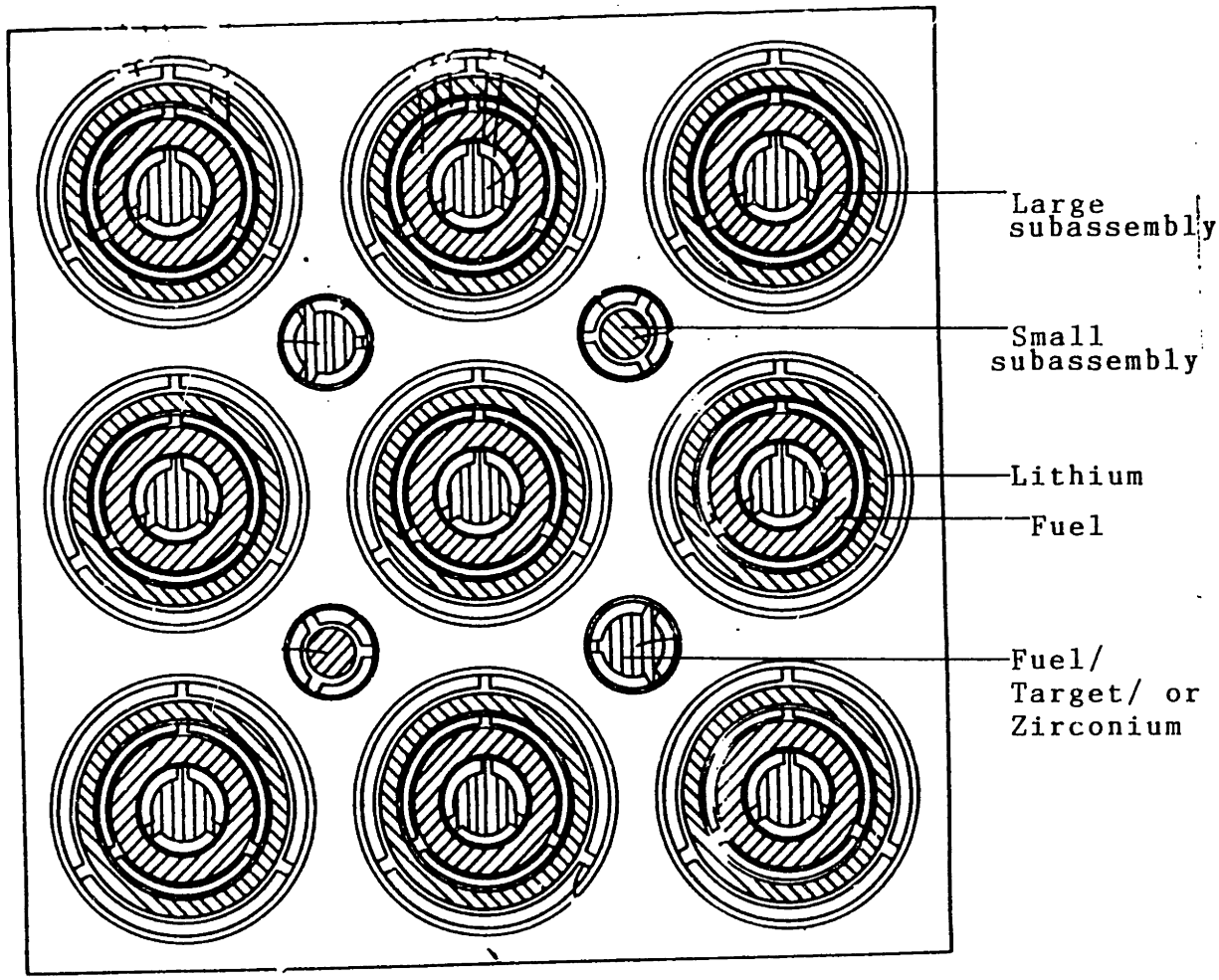
[13] William Cawley and Turner Trapp, U.S. patent 4,475,948, Oct 9, 1984 and U.S. patent 4,526,741, July 2 1985.



Fuel Assembly for Production of Tritium
In Light Water Reactors

Figure 6-4

6.3.3 Production in Fuel Rods and Control Guide Tubes

An assessment of the potential for the production of radioisotopes, e.g. Co^{60}, in various U.S. reactors was undertaken by Rupp, Cox, and Binford in the mid 1960s.[14] Although they did not specifically estimate potential tritium production, the similarity of the macroscopic absorption cross section of the enriched lithium aluminate/zirconium target material patented by Cawley et al and cobalt metal allows extrapolation to tritium production.[15]

Tritium Production Inferences

The Rupp et al production estimates shown in Table 6-1 were calculated under the assumption that 3% of the fuel

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[14] A.F. Rupp, J.A. Cox, F.T. Binford, "Radioisotope Production in Power Reactors," Oak Ridge National Laboratory Report number ORNL-3792, May 1965.

[15] In the Rupp et al study, it was assumed that the cobalt metal would be loaded in the reactor in that concentration that would produce a reduction in neutron flux by a factor of .8; the concentration of target material was adjusted accordingly.

The rate of neutron absorption in a given volume V, and a flux Ø is equal to ÊØV, where Ê is the macroscopic cross section of the material. Since the macroscopic cross section of both materials is approximately the same, the flux within a target will be the same, so the molar rate of Co60 and H3 will be roughly the same.

The average macroscopic absorption cross section of cobalt metal used in the Rupp calculation was (28.7x10^-24 cm^2)x(8.9g/cm^3 x 6.03 x10^23/59)=2.15 cm^-1;

The average macroscopic absorption cross section of the 10% volume lithium aluminate in the aluminate/zirconium mixture, assuming a Li6 enrichment near 100% is .1 x 2.55g/cm3 x 6x10^23/65 x 650x10^-24 = 1.53 cm^-1.

Table 6-1

Tritium Production in U.S. LWRs

Reactor Name	Reactor Type	Power (MWth)	Co production (grams)	Tritium production (grams)	Efficiency (milligrams/MWDth)
Shippingport	PWR	231	2204	114	1.69
Conn. Yankee	PWR	1473	4189	218	0.51
Oyster Creek	BWR	1600	7156	372	0.80
Yankee (Rowe, MA)	PWR	540	2714	141	0.89
Nine Mile Point	BWR	1538	4513	235	0.52
San Onofre 1	PWR	1210	3543	184	0.52

rods were replaced by target rods. The authors caution that "these estimates are only a first approximation. They are primarily for the purpose of gaining some idea of the production potential...." Thus, the inferred tritium production values should taken only as a rough approximation of the values that would be obtained by a detailed engineering study of each reactor.

To obtain production amounts for tritium, one must multiply the given values of Co^{60} by the ratio of the atomic weight of the tritium product to the atomic weight of the Co^{60} product (i.e. 3/60). The cobalt values must further be adjusted for differences in decay during irradiation, since Co^{60} has 5.24 year half life while tritium has 12.32 year half life. Calculations show that 6.3% of the total Co^{60} produced would decay during a one year irradiation, while only 2.7% of the tritium produced would decay.[16] Thus, the tritium equivalent of Co^{60} would be:

$$H^3=3/60 \times Co^{60} \times ((1-.027)/(1-.063)) = Co^{60} \times .052$$

where H^3 is the number of grams of tritium produced and Co^{60} is the number of grams of Co^{60} produced.

It should be noted from the table that there is no clear correlation between reactor type and production ability. It should also be noted that tritium production in power

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[16] The percent of an isotope that would decay during one year irradiation is equal to $1-(1/L)x(1-exp(-L)$, where L is the decay constant in years. A more general form of this equation was derived in section 4.3.

producing LWRs is on average .00082 g/MWD, while in production reactors it is roughly .0125 g/MWD.

In a 1000 MWe PWR, this production rate would correspond to roughly 720 grams of tritium produced per year by inserting lithium targets in roughly 1400 fuel tubes. If we assume that each rod produces an equal amount of tritium, an individual tube could be used to produce 0.5 grams of tritium[17]. Since control guide tubes have roughly the same dimensions as fuel tubes, the same quantity of tritium per tube could be produced by inserting lithium targets in empty control guide tubes.

The quantities calculated above represent an estimate of the maximum amount of tritium a power reactor could produce. They do not reflect any effort to hide production (e.g. they

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[17] A further estimate of potential tritium production in an individual tube can be calculated from the amount of boron that is consumed in a burnable poison rod in one year. An individual rod has a natural boron loading of 0.0603 g/cm (source: Lu, p. 2-30), with a total boron content of 22 grams (4.35 grams of B10). This entire quantity is consumed in one reactor cycle, typically one year. If the same number of moles of lithium were consumed, 1.3 grams of tritium would be produced. However, it is not clear that a reactor could operate safely for an entire cycle with a lithium target inserted in control or fuel tubes in a concentration high enough to match the borosilicate glass macroscopic cross section at the beginning of cycle. That is, although lithium could be inserted in the reactor in a concentration such that its macroscopic cross section matches that of borosilicate glass at the beginning of cycle, the cross sections of the two materials would decrease at different rates. Near the end of cycle, the macroscopic cross section of the borosilicate glass would be 0, while that of the lithium would still be very high. Nevertheless, lacking further detailed production computations, it may be advisable to use this higher number for safeguards calculations.

would require increased fuel enrichment, reduction in boron content in moderator water, etc.). Moreover, they do not reflect a detailed analysis of the operational difficulties associated with such production (e.g. flux depressions and power peaking). In section 6.3.4, below, tritium production is inferred from a study on plutonium production that takes into account operational difficulties as well as potentially detectable changes in reactor parameters.

6.3.4 Production Without Affecting Reactor Parameters

Lu, Zhu, and Todosow recently completed a study of unreported plutonium production in LWRs[18]. The study concentrated on the insertion of targets of natural uranium oxide in control guide tubes in PWRs. Targets would replace the thimble plugs and burnable poison rods.

Lu et al calculated that for a 3250 MWth PWR, 8 kilograms of plutonium could be produced by inserting 600 to 900 undeclared natural uranium oxide rods in 30 to 45 modified assemblies loaded in peripheral core locations. According to the authors:

> the presence of 32 modified assemblies in the peripheral locations in the reactor core... has a relatively mild impact on core parameters, and should not cause any operational difficulties. More importantly, the changes in soluble boron, cycle length, axial off-set (A.O.), and assembly powers for most locations are small, making it difficult to identify the presence of the modified assemblies.

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[18] M.-S. Lu, R.-B. Zhu, M. Todosow, "Unreported Plutonium Production in Light Water Reactors," Brookhaven National Laboratory report number ISPO-282, TSO 88-1, Feb. 1988.

Tritium Production Inferences

Comparison of tritium production to plutonium production suggests that between 35 and 115 grams of tritium could be produced in a PWR per year without safety problems or detectable effects on reactor operation.

Although a direct molar comparison between plutonium production and tritium production yields the 115 gram value, such a comparison is not as clearly applicable as is the previous comparison to cobalt production. The primary reason is that the neutron absorption by U238 in a PWR occurs largely by resonance absorption. That is, the neutrons absorbed by the uranium target typically have energies of several KeV. The neutrons absorbed by the lithium would be exclusively thermal. In a typical PWR, the ratio of neutrons consumed in U^{238} resonance absorption to those consumed in U^{238} thermal absorption is 3.23.[19] If lithium is inserted in the control guide tubes in a concentration such that its macroscopic absorption cross section matches the macroscopic thermal absorption cross section of uranium oxide,[20] roughly

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[19] Mason Benedict, Thomas Pigford, and Hans Wolfgang Levi, Nuclear Chemical Engineering, (New York: McGraw-Hill, 1981), p. 136.

[20] The macroscopic thermal absorption cross section of the natural UO2 target material is approximately 0.06 cm^-1. In order to match this cross section, a target material of unenriched lithium aluminate in a 1.6% volume concentration of the lithium aluminate/zirconium mixture described in

35 (i.e. 115 grams x 1/3.23) grams of tritium would be produced per year, without detectable effects on reactor operation.

6.3.4 Production Outside of Core

Areas outside of the core in PWRs provide additional locations for lithium irradiation. Regions of possible irradiation include space between the core baffle and core barrel; the core barrel and thermal shield; and the thermal shield and pressure vessel (see Figure 6-1).

Although these locations are convenient and would probably cause the least interference with the neutronics of the reactor core, the low flux levels in these area would demand large targets, which might affect cooling of the reactor. In any case, the insertion of lithium targets outside the core of a PWR could be easily detected by visual inspection. These areas are typically well lighted, and can be viewed during reactor refueling periods.[21]

6.4 Suggested Inspection Activities

As discussed above, diversion scenarios include: (1) altering fundamental fuel design; (2) replacing fuel with lithium; (3) inserting lithium in control guide tubes; and (4) placing lithium targets outside of the core. While

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section 6.3.1. could be used.

[21] Discussions with M. Todosow at Brookhaven National Laboratory on 11/12/88.

visual inspection of the core during refueling would allow the detection of (1) and (4), the detection of (2) and (3) would require detailed inspection of individual fuel rods, guide tubes and burnable poisons.

Such inspection would have to be accomplished at the fuel and control rod fabrication facilities. Inspection at these facilities would allow both destructive and non-destructive analysis of assembly materials and entire rods. Although use of a method that non-destructively tests entire fuel assemblies would be desirable, such methods are probably not sufficiently accurate to detect the insertion of small quantities of lithium in PWR assemblies. A typical PWR fuel fabrication facility produces roughly 1000 assemblies per year, enough for roughly 15 reactors.[22]

6.4.1 Inspection at Fabrication Facility

At a PWR fuel fabrication facility. UO2 power is received from the conversion plant, ground into a fine powder, and mixed if necessary with such burnable poisons as gadolinia oxide[23]. The powder is then compacted, shaped into pellets and sintered. The pellets are next loaded into zircaloy tubes that have been extruded, heat-treated, and

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[22] A commercial PWR fuel fabrication facility typically has an annual capacity of 500 MTHM. (source: Frank Rahn et al, A Guide to Nuclear Power Technology: A Resource for Decision Making, 1984, NY, Wiley., p. 874)

[23] Although the PWR described in section 6.1 utilized burnable poison rods separate from the fuel, other reactors may use a mixture of fuel and burnable poison.

ultrasonically inspected. End plugs are then welded to the top and bottom of the rod. Fuel rods are then assembled into fuel assemblies by insertion into a grid structure and packaged for shipment.[24]

Material in the production process would be sampled to ensure that the used burnable poison and control rod material is not lithium. Complete fuel rods would be non-destructively analyzed by active neutron interrogation. A fuel rod scanner that has been developed at Los Alamos for quality control tests at fabrication facilities could be used for this purpose.[25] The scanner, which consists of a rod conveyance system, standard nuclear electronic modules, and minicomputer hardware, operates by irradiating uranium fuel rods with moderated neutrons from a Cf-252 source. Fission product gamma rays are detected. The device measures uranium enrichment and the geometrical location of fissile material.

After approved fuel rods are assembled into an assembly, the entire assembly would be weighed to establish a base for comparison at the reactor, and then sealed. At the reactor, the seals and identification tags would be checked, and the assembly again weighed to ensure that targets have not been inserted in transit. Control assemblies would also be sealed

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[24] Frank Rahn, A Guide to Nuclear Power Technology: A Resource for Decision Making, (New York: Wiley, 1984), pp. 237-239.

[25] Clemens Auerbach, "Safeguards Instrumentation A Computer-Based Catalog, Second Edition," Brookhaven National Laboratory report number BNL-51450, April 1985, p.54.

after verification that they are not made of lithium.

A supplementary measure that might also be utilized is active neutron interrogation on the entire fuel assembly, both at the reactor and at the fabrication facility. The insertion of lithium in place of fuel, or in empty tubes would reduce the neutron multiplication rate from fission in an assembly of given fuel mass and enrichment. However, further study is required to determine if a device could be designed to detect the insertion of small quantities of lithium.

The IAEA currently uses a transportable interrogation system for LWR assemblies. An Am-Li neutron source, with spectrum moderated by polyethylene, actively interrogates a fuel assembly for fissile content with subthreshold thermal and epithermal neutrons. A large portion of the neutrons cause fission to occur. He3 detectors are used to detect the fission neutrons.[26]

A method of investigating assemblies where weight or reactivity measurements yield anomalous results might be the use of interstitial radiation probes in fresh fuel assemblies at the reactor. However, the feasibility of such a technique must be investigated, since no such device exists. Such an instrument might enable inspectors to distinguish between uranium fuel and lithium, as well as empty guide tubes and lithium filled tubes. The radiation probe would measure the

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[26] Auerbach, p. 53.

intensity of either the 186 kev gamma peak from U235, or the radiation from an inserted gamma radiation source.

Although interstitial probes are not currently used in IAEA safeguards applications, the machinery required to insert the probes into a fuel assembly has already been widely used in the fuel service industry to detect leaking rods via ultrasonic test. The machinery might be modified for safeguards application by substituting radiation probes for the ultrasonic transducers.[27]

6.4.2 Inspection Frequency

Inspection of every fuel element of every assembly, in every reactor in commercial operation would probably strain the resources of the verification agency and, as previously mentioned, might be excessively intrusive. The inspecting agency could, however pursue a random sampling strategy to detect lithium targets. It would be based on the two-level sampling strategy used by the IAEA in its verification activities. This strategy includes a large number of low accuracy measurements, and a small number of high accuracy measurements. It is designed to meet the dual challenge of the diversion of a large amount of nuclear material in a few number of fuel assemblies and the diversion of a small amount of material in a large number of assemblies.[28]

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[27] Lu, p. 7-5.

[28] Jonathan Sanborn, "Attributes Mole Sampling Schemes for International Material Accountancy Verification," Brookhaven National Laboratory, p. 1.

The fraction of fuel rods that would be sampled depends on the maximum feasible tritium production in an individual fuel assembly, as well as the accuracy of detection mechanism. As previously discussed, active neutron interrogation of complete fuel assemblies might constitute the lower accuracy measurement, and active interrogation of individual fuel rods might constitute the higher accuracy measurement.

6.5 LGRs and BWRs

The other widely used power reactor design in the U.S. is the BWR; in the Soviet Union it is the LGR. While the BWR poses a production threat similar to that of the PWR, the LGR poses a fundamentally different one. The primary distinguishing characteristic of this reactor is the absence of a pressure vessel which is replaced by pressure tubes, which allows refueling in an operating reactor.

6.5.1 Brief LGR Description

LGRs are moderated by graphite, but cooled by light water circulating through vertical channels in the core. The core (figure 6-3) of a typical 1000 MWe LGR is a vertical graphite cylinder with a diameter of 11.8 meters and a height of 7 meters. The graphite block that comprises the core is hermetically sealed. It contains 1661 process channels for fuel cartridges, and 211 control channels for neutron-absorbing control rods and monitoring system sensors. Water

is carried downward by individual pipes to each fuel channel.

In steady state operation, fuel is recharged at a rate of 1-2 cartridges per day by a refueling machine. The fresh fuel cartridge is set inside the process channel on a suspension support while the spent cartridge is removed.[29]

The fuel cartridges consist of two fuel element assemblies. The fuel element assemblies (Figure 6-4), in turn, consists of 18 rods of 2% enriched uranium oxide pellets. The average mass of fuel in a cartridge is 130 kg. The maximum power of an individual channel is 3250 kw. The burnup of fuel discharged from the reactor is approximately 20 MWD/kg.[30]

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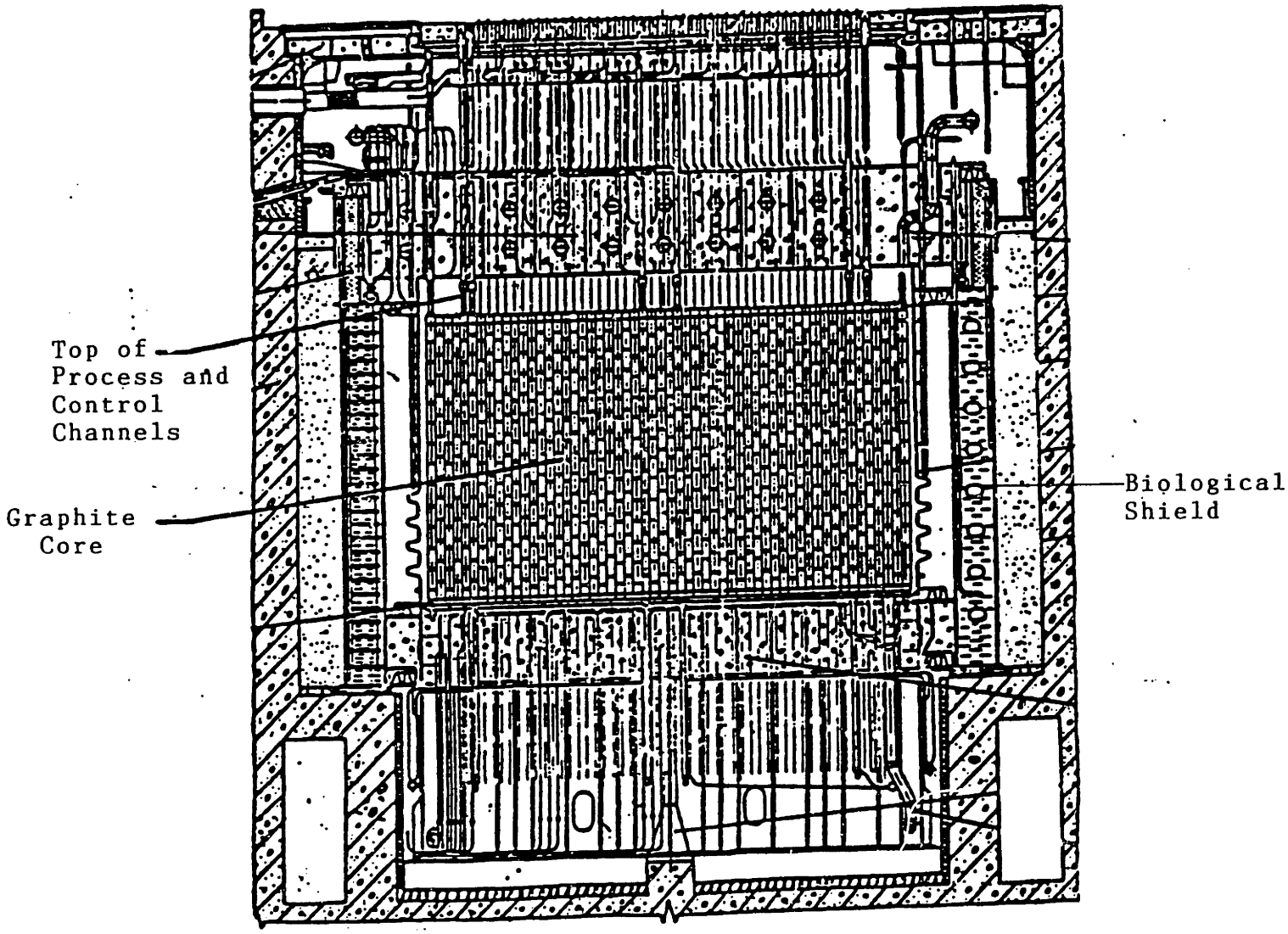
[29] State Committee for Using the Atomic Energy of the USSR, "The Accident at the Chernobyl AES and its Consequences," August 1986, translated by DOE, NR-40, pp.1-8.

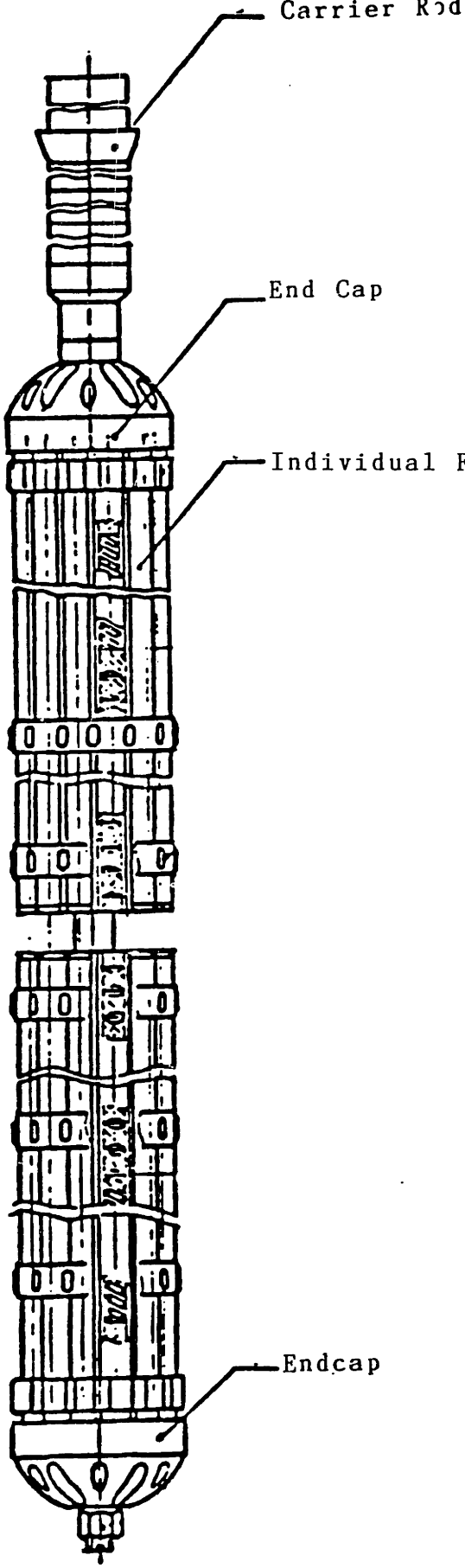
[30] S.C.U.A.E., U.S.S.R., p.32.



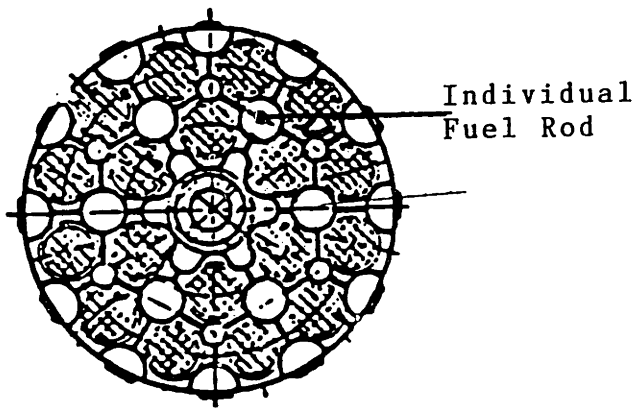
Cross Section of LGR Reactor

Figure 6-5

Source: U.S.S.R. State Committee for Using the Atomic Energy of the U.S.S.R., p. 2



LGR Fuel Element Assembly



Cross Section of LGR Assembly

Figure 6-6

Source: U.S.S.R. State Committee for Using the Atomic Energy of the U.S.S.R., p.6.

6.5.2 General Safeguards Considerations

Although it is not the goal of this analysis to present a safeguards approach for power reactors, two major safeguards related characteristics are apparent. First, since the reactor is refueled continually, insertion and removal of target rods could occur at any time during reactor operation. (In contrast, targets could be inserted in PWRs only during reactor shutdown, which occurs about once per year in normal operation) Second, the core of the LGR is not open to inspection, as is that of the PWR. However, this latter characteristic should not pose a grave safeguards problem as the reactor's core is normally hermetically sealed and would thus preclude insertion of targets in other than fuel and control guide positions.

Although the generic challenge presented by on-load fueled reactors is great, the IAEA has developed techniques and devices to safeguard CANDU on-load fueled reactors. Such safeguards make use of a bundle counting instrument, as well as TV surveillance systems, photosurveillance cameras, radiation monitors, and security seals. The surveillance cameras and radiation yes-no monitors verify that irradiated bundles are only removed from the core by the normal route (i.e. past the bundle counter to the spent fuel bay). The bundle counter determines the number of fuel bundles removed from the reactor and placed in the spent fuel bay. During inspection the inspector counts the number of bundles in the

spent fuel bay, and confirms that the count checks with bundle counter tallies.[31]

In the tritium safeguards approach for LGRs, a fuel cartridge (bundle) counter would also play a central role. However, the device would not operate in the same manner as the bundle counter for the CANDU reactor. The counting mechanism for the CANDU operates by detecting the gamma radiation emitted from spent fuel. However, clandestine lithium targets might not emit much gamma radiation, since tritium does not emit gamma radiation, and activation products in the lithium composition might be minimized by utilizing very pure target material. A mechanical device that monitors the operation of the fueling machine would be required. Such a device that uses electro-mechanical switches activated by the passage of a bundle is currently being tested by the IAEA for counting fresh fuel bundles inserted into the CANDU reactor[32]. The device would have to be altered to fit the fuel loading crane in the LGR.

In general, inspectors would determine that fuel and control assemblies do not contain lithium targets using techniques similar to those described in section 6.4, and apply identification seals. In the spent fuel bay, inspectors would check seals to ensure that assemblies are

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[31] Marvin Miller, "Heavy Water and Nonproliferation," MIT Energy Laboratory Report number, MIT-EL 80-009, May 1980, pp. 60-63.

[32] Aurbach, p. 82.

the same as those that have been certified. Surveillance cameras in the fresh fuel bay would be used to ensure that assemblies in that area are not tampered with. The cartridge (bundle) counter would allow inspectors to ensure that no more fuel cartridges were inserted in and removed from the core than were reported.

6.6 Other Reactors

Tritium production is also possible in research, demonstration, test, and training reactors. Such reactor types are even more varied than power reactors. Many are fueled by highly enriched uranium, and have less regular operating cycles than power reactors. The purpose of such reactors is to produce neutrons for basic research, the testing of reactor materials, production of radioisotopes, activation analysis, training, as well as to test reactor designs.

As with electric power producing reactors, tritium production capability in research reactors is determined by neutron flux and space in core for insertion of targets. Accessibility of space for target insertion is dependent on specific reactor design.

6.6.1 Low Powered Research Reactors

Low powered research reactors are typically utilized at universities and research institutes for basic physics experiments, production of small quantities of radioisotopes,

and materials testing. Reactors have circuits for irradiation of small samples and other ports for the introduction of targets into and around reactor core.

Eberhard, for example, has estimated that if each of the MIT 5 MW research reactor's three in-core sample assemblies were replaced with target assemblies of 10 kilograms of natural uranium, 100 g of Pu could be produced yearly[33], suggesting that 1.5 g of tritium could alternately be produced. Other estimates suggest that on the order of 1 to 2 kilograms of plutonium (15 to 30 grams of tritium) could be produced in a generic 10 MW materials test reactor dedicated to such production.[34]

Nevertheless, tritium production in many smaller research reactors may not be a verification threat. The average power of the 91 U.S. research reactors with individual power rating of less than 30 MW is 1.3 MW.[35] It is expected that a large portion of such smaller reactors could be exempt from very intrusive tritium safeguards. Periodic visual challenge inspections to verify that such reactors are not dedicated to tritium production may be

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[33] Carol Ann Eberhard, "A Case Study Application of IAEA Safeguards Assessment Methodology to the MIT Research Reactor," MIT Thesis, August 1982, p. 98.

[34] Marvin Miller, "The Potential for Upgrading Procedures at Research Reactors Fueled with Highly Enriched Uranium Part II," MIT, July 1984, p.4.

[35] Hatice Cullingford, "Alternatives to Proposed Replacement Production Reactors," Los Alamos National Laboratory report number LA-8867, June 1981, p. 9.

sufficient. An analysis of specific research reactor design must be undertaken.

6.6.2. Larger Research and Test Reactors

Larger research and commercial radioisotope production reactors pose a more formidable challenge. Cullingford calculates that:[36]

> -- the 40 MW High Flux Beam reactor at Brookhaven National Laboratory can produce 20 grams of tritium per year; and
> -- the 30 MW Oak Ridge Research Reactor can produce 30 grams of tritium.

Moreover, other large reactors such as the 62.5 MW Experimental Breeder Reactor (EBR-2) at Argonne National Laboratory and the 50 MW Loss of Fluid Test facility at Idaho National Engineering Laboratory have similar tritium production potentials.

6.7 Conclusion/Incentives for IAEA Participation

Although the tritium production capabilities of power reactors and large research reactors greatly exceed significant quantities, verification at civilian reactors appears feasible. However, inspection activities would be very intrusive and would require the measurement of individual fuel assemblies, and in some cases, individual fuel rods. Large research and test reactors may pose the

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[36] Cullingford, p. 9.

greatest challenge, as their operating mode is not as well defined as power reactors.

Although inspection of military facilities described in the previous three chapters would be accomplished primarily on a national basis, there are strong incentives for IAEA participation in verification at peaceful facilities. Current IAEA safeguards alone would clearly not suffice in verification of tritium non-production, and it is unlikely that either superpower would completely entrust verification to the Agency. However, such participation could lead to an increased international acceptance of IAEA safeguards - a goal that both superpowers support, while giving U.S. and U.S.S.R. verification agencies access to IAEA experience. Moreover, if the superpowers do not request IAEA assistance in verification, some countries might interpret this as an indictment of IAEA capabilities, thereby leading to reduced international acceptance of safeguards.

The United States, United Kingdom, France and the U.S.S.R have already concluded safeguards agreements with the IAEA, although they are not required to under the Nuclear Non-Proliferation Treaty. These countries have accepted safeguards on certain facilities to:

> -- address the concern of non-nuclear weapons states that nuclear weapons states have commercial advantage due expense and intrusion of safeguards implementation;
> -- give IAEA experience safeguarding facilities (such as uranium enrichment plants) that are not yet widely used by non-nuclear weapons states; and
> -- encourage adherence to the NPT and acceptance of IAEA safeguards.

The voluntary offers differ in scope. The U.S. and U.K. are the broadest, with application to all peaceful nuclear facilities.[37] However, even in the U.S., only a few facilities are currently safeguarded as widespread safeguarding would strain the resources of the Agency.[38] Nevertheless, U.S. and U.S.S.R. acceptance of these safeguards suggests that both powers recognize benefits in accepting IAEA safeguards.

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[37] Frank S. Houck, "The Voluntary Safeguards Offer of the United States," IAEA Bulletin 27, no. 2 (Summer 1985): p.13.

[38] Houck, p. 12.

CHAPTER SEVEN

CONCLUSIONS AND RECOMMENDATIONS

As noted at the outset of this study, a tritium limitation verification regime must include the following elements:

> (1) verification that tritium production limits are obeyed at allowed tritium production facilities;
> (2) verification that tritium is not produced in declared power and research reactors; and
> (3) verification that clandestine production facilities are not constructed or operated.

Because information on topic (3) is limited, the current analysis has concentrated on developing an approach for (1) and assessing the feasibility of (2).

The general conclusions of the analysis are as follows:

> -- A system to verify tritium production limits at a declared production facility can be developed with existing technology. Such a system would be based on the measurement of lithium depletion and tritium production. Measurement techniques for both lithium and tritium are available and have accuracies of approximately 1%;
>
> -- Power reactors and large research reactors are capable of producing tritium, although not as efficiently as production reactors. For example, a generic 1000MWe PWR is capable of producing roughly 1 kilogram of tritium if lithium targets are inserted in 1400 (3%) of its fuel or control guide tubes. However, such production might be detected by changes in reactor operation and increases in the enrichment of remaining fuel. A diverter could more confidently produce tritium in a PWR without detection by introducing fewer, less concentrated lithium targets into the control guide tubes. In this scenario, tritium production would be between 35 and 115 grams per year, and there would be no easily detectable

effects on reactor operation. Large research reactors, i.e. those whose thermal power is on the order of 30 MW, might be capable of producing roughly 20 grams of tritium per year if dedicated to such production.

- There is no fundamental technical reason why verification that significant quantities of tritium are not produced in peaceful nuclear facilities can not be accomplished by on-site inspection of fuel fabrication facilities and reactors. However, this inspection would be more intrusive than that provided for in any existing arms control treaty, and inspection at the fuel fabrication plant would probably be even more intrusive than the safeguards currently applied by the IAEA. Material at the fabrication facility would be sampled to ensure that the used burnable poison and control rod material is not lithium. Complete fuel rods would be non-destructively analyzed by active neutron interrogation. A fuel rod scanner that has been designed for quality control tests at fabrication facilities could be used for this purpose. After approved fuel rods are assembled into the assembly, the entire assembly would be weighed, and sealed. At the reactor, the seals would be checked, and the assembly again weighed to ensure that targets have not been inserted in transit. Soviet LGRs present a formidable safeguards challenge as they are fueled on-load. However, techniques have been developed for on-load reactors, and they could be applied to LGRs with some modification. Large research reactors may require the most intrusive verification as their operation schedule may not be as well defined as that of power reactors.

7.1 Further Research

In addition to an in depth analysis of national capabilities to detect clandestine tritium production facilities, other areas of required further research can be identified from the current analysis. These areas are:

-- development of a detailed safeguard approach for power reactors;
-- development of specific activities for verification of tritium in peaceful uses;

-- investigation of the feasibility of clandestine production in sea-based reactors;
-- development of safeguards techniques for extraction of tritium from reactor heavy water;
-- development of techniques for verification at radioisotope reactors; and possibly
-- development of a plan for verifying tritium in storage.

Each of these areas is briefly discussed below.

7.1.1 Power Rectors

The most pressing requirement for further research is the development of a detailed safeguards approach for civilian nuclear reactors. The current analysis has given estimates of potential clandestine production at PWRs and research reactors, and suggested techniques for detecting the presence of lithium targets in such reactors. However, a detailed safeguards approach for specific reactors needs to be developed.

7.1.2 Peaceful Applications

As mentioned in section 3.2.2, over one significant quantity of tritium is supplied to U.S. and foreign companies and institutes for commercial application each year. For example, in 1986, 62 grams of tritium were supplied for domestic non-military use and 117 grams were supplied for foreign non-military use. The key issue in the area of peaceful uses is ability of a treaty party to assemble a significant quantity from dispersed sources. Procedures to ensure that these supplies of tritium are not diverted for military application must be developed.

7.1.3 Heavy Water Reactors

In heavy water moderated power reactors tritium is formed through neutron bombardment of deuterium. The following reaction occurs with a 2200 m/sec cross section of 5.7×10^{-4} barn.

$$H^2 + n^1 \text{---} > H^3$$

For a 1000 MWe CANDU-type power plant, the yearly production rate of tritium in the heavy water is approximately 200 grams at 80% capacity.[1]

Although no facilities for the separation of tritium from heavy water are operated in the U.S., two such facilities are operating in Canada, and one in France[2]. Although the U.S. and U.S.S.R. do not operate heavy water power reactors, the SRL reactors are moderated by heavy water, and stockpiles of tritiated heavy water are stored at SRL. A safeguards approach involving sampling the heavy water to assay the tritium content must be developed to guard against diversion of these stockpiles.

7.1.4 Radioisotope Reactors

Various radioisotopes are produced in both the U.S. and the U.S.S.R. for peaceful applications, e.g. Co^{603}. They are

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[1] Marvin Miller, "Technology to Extract Tritium from Heavy Water," Department of Nuclear Engineering, M.I.T., October 1987 (revised), p. 2.

[2] Miller, pp. 4-5.

[3] "List of DOE Radioisotope Customers with Summary of Radioisotope Shipments, FY 1986," report number PNL-5948, 1987, p. 5.1-5.4.

produced in basically the same way as is tritium. Target material is inserted into a reactor and irradiated. Verification activities could include the detection of the radiation and heat emitted by the produced radioisotopes. Detailed procedures for such verification must be developed.

7.1.5 Sea Based Reactors

The current analysis has considered only land-based production facilities and power reactors. Sea-based reactors could also be used to produce tritium. Although the reactors that power nuclear submarines are not optimal in this regard due to the lack of space in and around the cores of such reactors, as well as their infrequent refueling, production potential on nuclear powered surface ships, such as nuclear powered ice breakers, must be investigated.

7.1.6 Tritium Storage

It is clear that, in at least the first several years of treaty implementation, no tritium production would be required. That is, tritium removed from existing weapons could be used to compensate for decay in weapons. For example, for a 50% reduction in arms, the 40 kilograms removed from the current U.S. stockpile could presumably be used to replenish existing weapons for roughly 12 years. It may be desirable to place the tritium from disassembled weapons in a verifiable storage regime. Tritium could be stored in the form of tritides in a central facility. Storage techniques are discussed in chapter three. Specific

verification procedures must be developed.

7.2 Future Challenges

This paper has analyzed existing tritium production technologies. Two technical challenges to the regime may emerge in the next several decades. These challenges are:

-- the advent of fusion power reactors; and
-- the construction of large accelerators;

7.2.1 Fusion Power

The advent of fusion energy will introduce severe verification complications to verification regime. As mentioned in section 3.2, a 1000MWe plant of the currently considered Tokamak design will produce 562 grams of tritium per day, and will consume 536 grams per day. Moreover, it will contain roughly 11 kilograms of tritium in inventory. Clearly, the handling and storage of such large amounts of tritium would introduce a severe verification challenge to the regime. If fusion power becomes a commercial reality, verification procedures for such reactors will need to be developed.

7.2.2 Large Accelerators

Linear accelerators that produce proton or deuteron particle beams with energies above 800 MeV could be designed to produce a shower of neutrons for the irradiation of lithium targets. The charged particles would strike a target, such as lead, and high energy neutrons would be

produced spallation. Although no existing LINAC has a particle current sufficient to produce significant quantities of tritium, such accelerators may be constructed in the future. An accelerator designed to produce tritium would be easily identifiable as such due to cooling requirements in the target.[4]

7.3 A Closing Note on Tritium Control

The analysis indicates that an arms control agreement that limits the production of tritium at declared production facilities and power reactors could probably be verified, although inspections of unprecedented intrusiveness would be required. However, the fact that an arms control measure is verifiable does not mean that it is necessarily a good idea. As discussed in chapter one, nuclear materials control is useful in that it can help to preclude the development of new weapons and can put an upper limit on the number of existing weapons. The fundamental objectives of current arms control efforts are to enhance deterrence and reduce the possibility of accidental nuclear war. Nuclear material control efforts must be analyzed by strategic planners within this context.

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[4] Warren Stern, "Memo to Marvin Miller: Tritium Production Potential of Linear Accelerators," September, 1988. The memo shows that the most powerful existing LINAC at LANL could produce, under very generous assumptions, no more that 20 grams of tritium per year. Moreover such production would require a fundamental redesign of the accelerator target facility.

Appendix A

Fundamentals of Mass Spectrometry[1]

Mass spectrometry is a widely used form of destructive material analysis and is the suggested measurement technique in this study. The following brief discussion provides an introduction to the general design and operation of mass spectrometers.

A mass spectrometer produces a beam of gaseous ions from a sample material; sorts out the resulting mixture of ions according to their mass-to-charge ratios; and provides output signals, which are measures of the relative abundance of each species present. Mass spectrometers are classified on the basis of how mass separation is accomplished.

Every mass spectrometer includes four components: (1) a source, where a beam of sample ions is generated; (2) an analyzer, in which separation of the beam ions is accomplished (either in space or in time); (3) a detector, where the separated ions are detected and their intensity measured (the mass to charge ratio reveals what is present, and the measured intensities reveals how much is present); and (4) a vacuum system, which provides the required environment for the previous processes. Figure A-1 shows

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[1] Appendix A is derived primarily from Introduction to Mass Spectrometry Instrumentation And Techniques, John Roboz, (New York: John Wiley & Sons, 1968).

the main features of a mass spectrometer system.

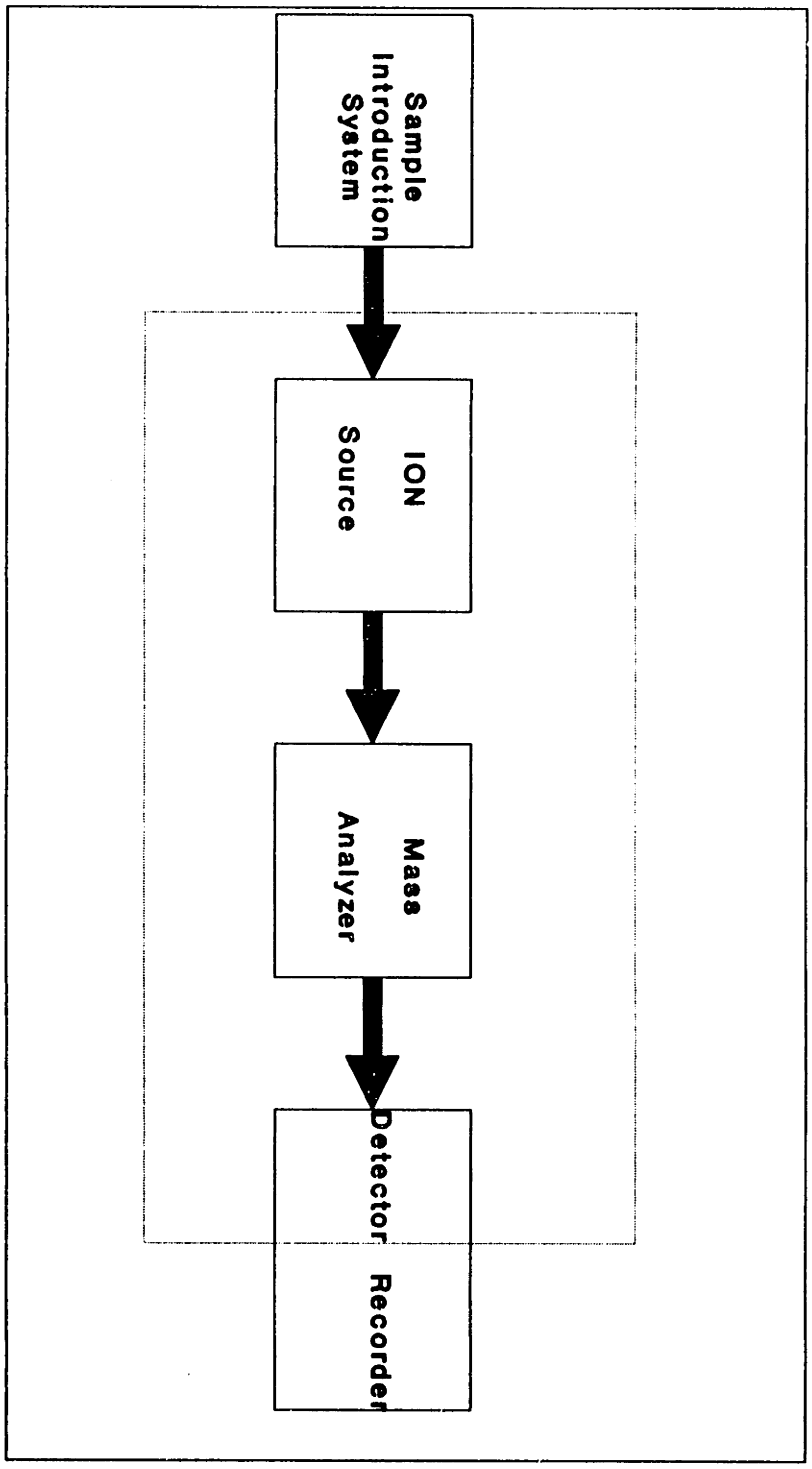

Figure A-1 Diagram of Mass Spectrometer System

Source

The ion source produces a beam of ions from the neutral sample. There are many techniques for such ion production. The choice of the particular source is dictated by the character of the sample to be analyzed (e.g. solid, gas, quantity, etc.) and the type of information sought (coverage, sensitivity required, bulk vs surface analysis, etc.). The purpose of an ion source is to produce a beam of ions which accurately represents the sample. Ion sources have the following primary characteristics:

-- energy spread;
-- sensitivity;
-- ionic species produced;
-- background and memory; and
-- ion current stability and noise.

The most important characteristic is probably the energy spread, since this directly affects the type of mass analyzer required (single or double-focusing).

The most widely used method of ion production in mass spectrometry is based on collisions between neutral atoms or molecules and energetic electrons (approximately 70 eV) at low pressure (10^-4 to 10^-6 torr). This is called the **electron bombardment source.** (Almost all organic mass spectrometry is accomplished with electron bombardment source). Two other types of sources are **field emission source,** in which ionization is achieved by strong electrical **fields; and photoionization source** in which ultraviolet light **is employed.**

Inorganic solids, are usually ionized by either thermal ionization sources, where ionization is accomplished by evaporation from a heated metal surface; or in a vacuum discharge source, where ions are formed in sparks or arks. The thermal ionization source is based on the fact that when neutral atoms or molecules are heated on, or impinge on, a hot metallic surface, there is a probability that ions will evaporate in addition to neutral particles. The mass spectrometer for lithium analysis described in chapter 4 uses a thermionic ion source.

Mass Analyzer

Mass analyzers are classified as either static type analyzers or dynamic type analyzers. The former utilize the momentum dispersion properties of magnetic fields and the energy dispersion properties of electric fields to accomplish separation. The latter are based on the time dependence of a parameter in a system such as the time of flight of ions in evacuated tubes, or the time dispersion properties of the radio frequency field. Both of the mass spectrometers suggested for tritium production verification are static type analyzers.

In a typical static analyzer, an electrostatic field is used to accelerate ions to equal energies. The beam of ions is then passed through a perpendicular magnetic field of strength B. The deflection of the ions will be described by

a circular trajectory of radius r, determined from the following equation:

(1) $$m/e = B^2 r^2 / 2V$$, where

where:
m is the mass of the particle
e is the charge
B is magnetic flux density
V is acceleration voltage

A mass spectrometer using an electrostatic accelerator to obtain an ion beam of homogenous energy, and a homogenous magnetic field to effect mass dispersion is illustrated in figure A-2. The magnetic field is normal to the plane of the figure.

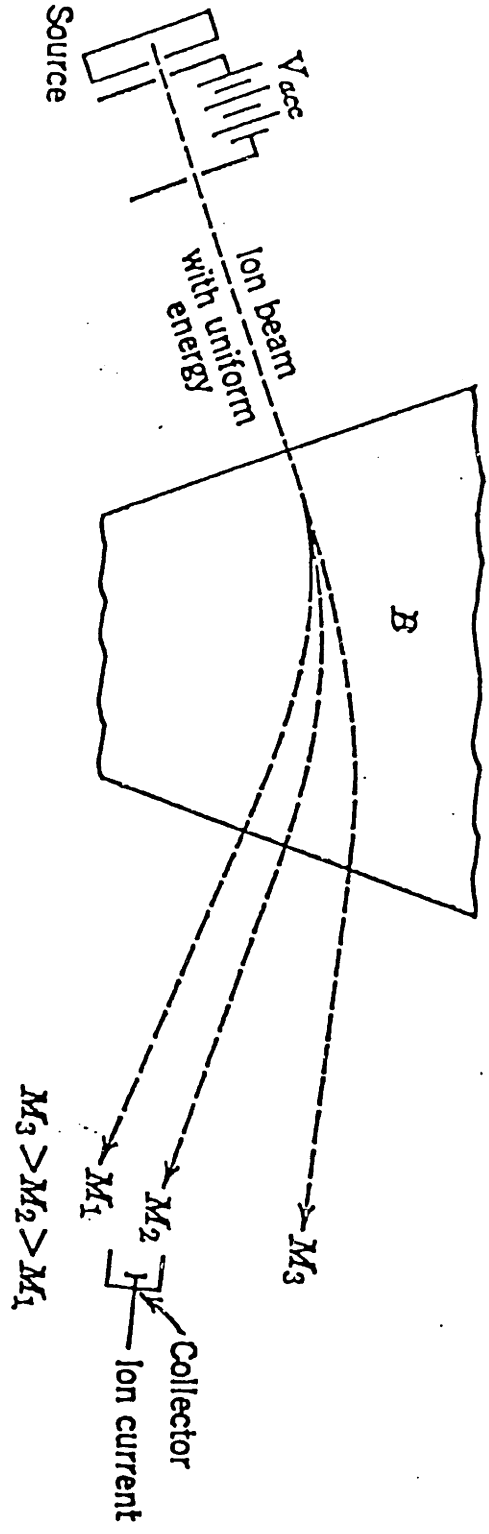

Figure A-2 Action of a homogeneous magnetic field on ions of equal energy, but different mass.

The process whereby ions are brought onto the receiving collector is called scanning. As shown by equation (1), scanning can be accomplished by either varying V for a fixed B, or varying B, for a fixed V. Increasing the magnetic field while keeping other variables constant, or alternatively decreasing the acceleration voltage, while keeping the other variables constant, will focus heavier and heavier ions on the collector. As mentioned in chapter four, the mass spectrometer for tritium analysis can analyze a sample in five minutes by magnetic scanning and in 2 minutes by voltage scanning. The spectrometer for lithium analysis scans by varying the magnetic field.

The resolution of a mass spectrometer is a measure of its ability to separate and identify ion that have nearly the same masses.

Detectors

Ion detectors are based either on the conversion of the individual separated ion beams into a proportional electron current which can be amplified and recorded, or on the simultaneous collection of all ion beams using ion sensitive photoemulsions. The defining characteristics of detectors are their sensitivity and speed of response.

In conventional detectors, the positive ions arriving at the collector are neutralized by electrons arriving from ground after passing through a high ohmic resistor. The potential drop across the resistor is the measure of the ion

current. Amplification of the potential drop is facilitated by a direct current amplifier or a vibrating reed electrometer.

APPENDIX B

U.S. AND SOVIET POWER REACTORS IN COMMERCIAL OPERATION

U.S.:

Reactor	Location	MWe	Type	Date of Commercial Operation
NORTHEAST				
Calvert Cliffs 1	Lusby, MD	850	PWR	5/75
Calvert Cliffs 2	Lusby, MD	850	PWR	4/77
Pilgrim 1	Plymouth, MA	670	BWR	12/72
Haddam Neck	Haddam Neck, CT	582	PWR	1/68
Indian Point 2	Indian Point, NY	873	PWR	7/74
Beaver Valley 1	Shippingport, PA	833	PWR	4/77
Beaver Valley 2	Shippingport, PA	833	PWR	11/87
Oyster Creek 1	Forked River, NJ	620	BWR	12/69
Three Mile Island 1	Londonderry Twp., PA	792	PWR	9/74
Maine Yankee	Wiscasset, ME	825	PWR	12/72
Indian Point 3	Indian Point, NY	965	PWR	8/76
James A. Fitzpatrick	Scriba, NY	821	BWR	7/75
Nine Mile Point 1	Scriba, NY	610	BWR	12/69
Nine Mile Point 2	Scriba, NY	1080	BWR	3/88
Millstone 1	Waterford, CT	660	BWR	12/70
Millstone 2	Waterford, CT	870	PWR	12/75
Millstone 3	Waterford, CT	1150	PWR	4/86
Susquehanna 1	Berwick, PA	1050	BWR	6/83
Susquehanna 2	Berwick, PA	1050	BWR	2/85

(Source: "World List of Nuclear Power Plants," Nuclear News 31, no. 10 (August, 1988): pp.83-87.)

Peach Bottom 2	Peach Bottom, PA	1065	BWR	7/74
Peach Bottom 3	Peach Bottom, PA	1065	BWR	12/74
Limerick 1	Pottstown, PA	1055	BWR	2/86
Salem 1	Salem, NJ	1106	PWR	6/77
Salem 2	Salem, NJ	1106	PWR	10/81
Hope Creek 1	Salem, NJ	1067	BWR	2/87
Robert E. Ginna	Ontario, NY	490	PWR	3/70
Vermont Yankee	Vernon, VT	514	BWR	11/72
Yankee	Rowe, MA	175	PWR	6/61
MIDWEST				
Perry 1	North Perry, OH	1205	BWR	11/87
Dresden 2	Morris, IL	794	BWR	8/70
Dresden 3	Morris, IL	794	BWR	10/71
Lasalle County 1	Senaca, IL	1078	BWR	10/82
Lasalle County 2	Senaca, IL	1078	BWR	6/84
Clinton 1	Clinton, IL	930	BWR	4/87
Zion 1	Zion, IL	1040	PWR	12/73
Zion 2	Zion, IL	1040	PWR	9/74
Byron 1	Byron, IL	1120	PWR	9/85
Byron 2	Byron, IL	1120	PWR	8/87
Quad-Cities 1	Cordova, IL	789	BWR	8/72
Quad-Cities 2	Cordova, IL	789	BWR	10/72
Big Rock Point	Charlevoix, MI	69	BWR	12/62
Fermi 2	Newport, MI	1093	BWR	1/88
Palisades	South Haven, MI	777	PWR	12/71
Donald C. Cook 1	Bridgman, MI	1020	PWR	8/75
Donald C. Cook 2	Bridgman, MI	1060	PWR	7/78
Duane Arnold	Palo, IA	538	BWR	2/75
Cooper	Brownsville, NE	778	BWR	7/74
Monticello	Monticello, MN	536	BWR	7/71
Prairie Island 1	Red Wing, MN	520	PWR	12/73
Prairie Island 2	Red Wing, MN	520	PWR	12/74

Fort Calhoun 1	Fort Calhoun, NE	486	PWR	9/73
Davis-Besse 1	Oak Harbor, OH	866	PWR	11/77
Callaway 1	Fulton, MO	1150	PWR	4/85
Point Beach 1	Two Creeks, WI	485	PWR	12/70
Point Beach 2	Two Creeks, WI	485	PWR	10/72
Kewaunee	Carlton, WI	535	PWR	6/74
Wolf Creek	Burlington, KS	1150	PWR	9/85
SOUTH				
Joseph M. Farley 1	Dothan, AL	829	PWR	12/77
Joseph M. Farley 2	Dothan, AL	829	PWR	7/81
Nuclear One 1	Russellville, AK	836	PWR	12/74
Nuclear One 2	Russellville, AK	858	PWR	3/80
Robinson 2	Hartsville, SC	665	PWR	3/71
Brunswick 1	Southport, NC	790	BWR	3/77
Brunswick 2	Southport, NC	790	BWR	11/75
Shearon Harris	New Hill, NC	900	PWR	5/87
Oconee 1	Seneca, SC	860	PWR	7/73
Oconee 2	Seneca, SC	860	PWR	9/74
Oconee 3	Seneca, SC	860	PWR	12/74
Mcguire 1	Cornelius, NC	1180	PWR	12/81
Mcguire 2	Cornelius, NC	1180	PWR	3/84
Catawba 1	Clover, SC	1145	PWR	6/85
Catawba 2	Clover, SC	1145	PWR	8/86
Turkey Point 3	Florida City, FL	666	PWR	12/72
Turkey Point 4	Florida City, FL	666	PWR	9/73
St. Lucie 1	Hutchinson Island, FL	827	PWR	12/76
St. Lucie 2	Hutchinson Island, FL	837	PWR	8/83
Crystal River 3	Red Level, FL	825	PWR	3/77
Edwin I. Hatch 1	Baxley, GA	810	BWR	12/75
Edwin I. Hatch 2	Baxley, GA	820	BWR	8/79

Vogtle 1	Waynesboro, GA	1100	PWR	5/87
River Bend 1	St. Francisville, LA	940	BWR	6/86
Waterford 3	Taft, LA	1104	PWR	9/85
Virgil C. Summer 1	Parr, SC	900	PWR	1/84
Grand Gulf 1	Port Gibson, MS	1250	BWR	7/85
Browns Ferry 1	Decatur, AL	1067	BWR	8/74
Browns Ferry 2	Decatur, AL	1067	BWR	3/75
Browns Ferry 3	Decatur, AL	1067	BWR	3/77
Sequoyah 1	Soddy-Daisy, TN	1148	PWR	7/81
Sequoyah 2	Soddy-Daisy, TN	1148	PWR	6/82
Surry 1	Gravel Neck, VA	781	PWR	12/72
Surry 2	Gravel Neck, VA	781	PWR	5/73
North Anna 1	Mineral, VA	893	PWR	6/78
North Anna 2	Mineral, VA	893	PWR	12/80
SOUTHWEST				
Palo Verde 1	Wintersburg, AZ	1221	PWR	1/86
Palo Verde 2	Wintersburg, AZ	1221	PWR	9/86
Palo Verde 3	Wintersburg, AZ	1221	PWR	1/88
WEST AND NORTHWEST				
Diablo Canyon 1	Avila Beach, CA	1084	PWR	5/85
Diablo Canyon 2	Avila Beach, CA	1106	PWR	3/86
Trojan	Prescott, OR	1130	PWR	5/76
Fort St. Vrain	Platteville, CO	330	HTGR	1/79
Rancho Seco	Clay Station, CA	913	PWR	4/75
San Onofre 1	San Clemente, CA	436	PWR	1/68
San Onofre 2	San Clemente, CA	1100	PWR	8/83
San Onofre 3	San Clemente, CA	1100	PWR	4/84
WNP-2	Richland, WA	1100	BWR	12/84

U.S.S.R.:

Ministry of Medium Machine Building

Troitsk A	Troitsk	100	LGR	9/58
Troitsk B	Troitsk	100	LGR	12/59
Troitsk C	Troitsk	100	LGR	12/60
Troitsk D	Troitsk	100	LGR	12/61
Troitsk E	Troitsk	100	LGR	12/62
Troitsk F	Troitsk	100	LGR	12/63

Ministry of Nuclear Power

Beloyarskiy 2	Zarechnyy	175	LGR	12/67
Beloyarskiy 3	Zarechnyy	550	LMFBR	10/80
Novovoronezhskiy 1	Novovoronezhskiy	265	PWR	12/64
Novovoronezhskiy 2	Novovoronezhskiy	338	PWR	4/70
Novovoronezhskiy 3	Novovoronezhskiy	410	PWR	6/72
Novovoronezhskiy 4	Novovoronezhskiy	410	PWR	4/73
Novovoronezhskiy 5	Novovoronezhskiy	950	PWR	7/81
Kola 1	Polyarnyye Zori	440	PWR	12/73
Kola 2	Polyarnyye Zori	440	PWR	12/74
Kola 3	Polyarnyye Zori	440	PWR	12/82
Kola 4	Polyarnyye Zori	440	PWR	11/84
Armenia 1	Metsamor	400	PWR	10/77
Armenia 2	Metsamor	400	PWR	5/80
Leningrad 1	Sosnovyy Bor	1000	LGR	12/74
Leningrad 2	Sosnovyy Bor	1000	LGR	9/75
Leningrad 3	Sosnovyy Bor	1000	LGR	6/80
Leningrad 4	Sosnovyy Bor	1000	LGR	10/81
Kursk 1	Kurchatov	950	LGR	12/76

Kursk 2	Kurchatov	950	LGR	6/79
Kursk 3	Kurchatov	950	LGR	12/83
Kursk 4	Kurchatov	950	LGR	12/85
Rovno 1	Kuznetsovsk	440	PWR	6/82
Rovno 2	Kuznetsovsk	440	PWR	12/82
Rovno 3	Kuznetsovsk	950	PWR	12/86
South Ukraine 1	Konstantinovka	950	PWR	6/84
South Ukraine 2	Konstantinovka	950	PWR	12/84
South Ukraine 3	Konstantinovka	950	PWR	/87
Smolensk 1	Desnogorsk	950	LGR	7/83
Smolensk 2	Desnogorsk	950	LGR	5/85
Chernobyl 1	Pripyat	950	LGR	5/78
Chernobyl 2	Pripyat	950	LGR	5/79
Chernobyl 3	Pripyat	950	LGR	6/82
Kalinin 1	Udomlya	950	PWR	12/84
Kalinin 2	Udomlya	950	PWR	12/86
Zaporozhye 1	Energodar	950	PWR	11/84
Zaporozhye 2	Energodar	950	PWR	6/85
Zaporozhye 3	Energodar	950	PWR	/87
Zaporozhye 4	Energodar	950	PWR	12/87
Ignalina 1	Snieckus	1450	LGR	12/84
Ignalina 2	Snieckus	1450	LGR	/87
Khmel'nitskiy 1	Neteshin	950	PWR	12/87
Balakovo 1	Balakovo	950	PWR	6/86
Balakovo 2	Balakovo	950	PWR	10/87
Scientific Research Institute for Atomic Reactors				
VK-50	Dimitrovgrad	50	BWR	1/66
State Committee on Atomic Energy				
BN-350	Shevchenko	350	LMFBR	7/73

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