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FUSION SAFETY

PROCEEDINGS OF A WORKSHOP
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN VIENNA, 23–27 MARCH 1981

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FOREWORD

Nuclear fusion could become a viable energy source. Work in plasma physics and fusion research is progressing rapidly in various Member States and international collaboration continues on work aimed at the demonstration of fusion power generation. Environmental protection and safety programmes need to keep pace with the present technological developments of fusion research. In order to provide a forum for discussing safety questions, including environmental safety aspects, related to fusion power development, the IAEA had organized a Technical Committee entitled "Workshop on Fusion Safety". It convened during the period 23-27 March 1981 at the Agency's Headquarters in Vienna. This volume contains the proceedings of this workshop. It is hoped that the information provided will contribute to a better understanding of the existing international data base in the field of fusion safety.

CONTENTS

1. INTRODUCTION

1.1 Background, objectives and scope.

In November 1979 an IAEA Consultants' Meeting was held to consider environmental and safety aspects of fusion. The main recommendation of the consultants was that an IAEA workshop on fusion safety should be held during the year 1981. The International Fusion Research Council (IFRC) endorsed this recommendation and agreed to the proposed terms of reference for the workshop. The workshop should achieve:

i) an improved international understanding of fusion reactor safety,
ii) better understanding of the international data base available to address fusion safety,
iii) co-ordination of current, similar on-going safety related programmes,
iv) consensus on the key areas where additional research or analysis is needed and the relative priorities of the various tasks,
v) promotion of conduct of fusion safety research on a broader international front.

Topics suggested for the workshop were:

i) tritium handling and safety
ii) activation product generation and release
iii) lithium safety
iv) superconducting magnet safety
v) potential fusion reactor accidents
vi) safety analysis techniques
vii) operational safety
viii) shielding considerations
ix) environmental impacts
x) waste management considerations and techniques; decommissioning
xi) biological effects of magnet fields

1.2 Organization and Participation

The workshop was held during the period 23-27 March 1981 at the Agency's Headquarters in Vienna, Austria. It was attended by 20 experts from 7 Member States and two international organizations, the CEC and the IAEA. A list with the names of the experts and their affiliation is included in these proceedings.

1.3 Workshop Activities

Experts from the USA, the UK and France presented overview statements of fusion safety activities in progress in their respective countries. Summaries of these statements are included in these proceedings. The Scientific Secretary of the workshop summarized the history of the IAEA's involvement in fusion safety. 7 papers were presented. The full text of these papers together with the questions and answers to the papers is included in these proceedings.

The workshop participants discussed fusion safety research and development requirements in the suggested topic areas with the intention of identifying outstanding problems that should receive highest priority of attention within the near future. The list of high priority fusion safety research and development needs that resulted from this discussion is also included in these proceedings.

The workshop participants agreed on recommendations to the IAEA on procedures to further promote, co-ordinate and publicize future fusion safety work on an international basis.

One session of the workshop was devoted to a discussion of current safety issues of INTOR. As the safety work specifically done for INTOR will be reflected in the IAEA's publications of the INTOR workshop, the presentations of the INTOR related safety studies were not incorporated in the proceedings of this workshop.

1.4 General Conclusion

There is a growing involvement in fusion safety or safety related programmes in Member States of the IAEA, mostly related to the present and next generation of fusion experiments but also arising from the range of conceptual fusion reactor designs. Dedicated R & D programmes on safety topics, e.g. tritium handling, lithium fires and magnet safety, are being supported, particularly in the U.S.A. which has a comprehensive centrally co-ordinated safety programme.

Further programme expansion is required on both short and long term basis. There is an urgent need to up-date existing reviews on fusion safety and to establish procedures for close international collaboration and co-ordination in this subject area. In particular, it is desirable to establish a reliable data background as a base for an environmental impact statement of fusion.

The Workshop included delegates, specialists in some of the safety topic areas, as well as others with wider general involvement. It was considered realistic to provide a consensus view on problems and priorities on many safety topics but additional information and analysis was needed from other specialist individuals and groups.

STATEMENTS ON NATIONAL FUSION SAFETY PROGRAMMES

SUMMARIES

FRANCE

J.L. Rouyer

Several teams of the French "Commissariat à l'Energie Atomique" (C.E.A.) are involved in fusion studies. It is only recently that the "Institut de Protection et Sureté Nucléaire" (I.P.S.N.) has been involved in fusion safety and protection activities. This contribution is essential in the actual technological phase of fusion development. That is why I.P.S.N. is participating in a Group of Experts to study fusion issues with physicists, metallurgists, chemists. It is hoped that a safety research programme specific to fusion be developed.

Work is also going on in co-operation with the Chemistry Division of CEA for the specification of the JET tritium plant, including clean-up, ventilation and monitoring systems.

UNITED KINGDOM

T.K. Allen

The United Kingdom involvement in fusion safety has developed in parallel with the fusion research programme, initially through conventional safety and reliability requirements of magnetic containment experiments and later, 1967 - 75, extending to the environmental aspects of fusion reactors as an integral feature of conceptual reactor design studies. On-going consideration of reactor design give high priority to safety aspects and relevant expertise from the UK nuclear programme has been mobilised in the assessment of radiation, shielding, activation, remote maintenance and waste management areas. As yet there has been no large scale R&D programme specific to fusion safety.

The JET project, sited at Culham, has more recently focussed attention on the radiation and environmental aspects of the present generation of large tokamak experimental programmes. JET, the UKAEA and other European organisations have combined to provide information on the neutron flux during D-T operation and on the required shielding and operational procedures. Studies on tritium handling and radiological protection are well advanced. Preliminary studies of waste management, de-commissioning and disposal are under way.

UNITED STATES OF AMERICA

G.R. Nardella

The United States Fusion Safety Programme has been designed to maintain pace with overall magnetic fusion development programme and with construction and operation of the Fusion Engineering Device (FED). In order to help meet this objective DOE had designated Idaho National Engineering Laboratory (INEL) as lead laboratory in Fusion Safety Research and in accordance with implementing its role as lead laboratory, INEL developed a Fusion Safety Research Programme Plan which will guide the safety work over the next several years. Generally the safety programme supports work in three areas and they are

1) identification and control of the radioactive inventories in fusion facilities.

   The programmes in this area include construction and operation in 1982 of the Tritium Systems Test Assembly (TSTA) at LANL which will develop and demonstrate the operation of a prototypic fusion reactor fuel cycle system and control and containment of reactor relevant quantities of tritium. This facility will also conduct tritium research in several areas including adsorption and desorption mechanisms, permeation and development of effective cleanup and decontamination techniques. Besides the work at TSTA there is also additional tritium research being conducted in the area of barrier development, probabilistic risk and modeling assessment and a determination of the biological effects of tritium. Analytical tools are also being developed to determine the consequences of releases of activated products including identification and assessment of potential disruptive energy sources and the transport and deposition of products within fusion reactor coolant system. Also a determination of the feasibility of using low activated material to improve maintenance and waste management aspects of a fusion reactor is being done.

2) identification and mitigation of the energy sources potentially available to release a portion of these inventories.

   The programmes in this area focus on examining the safety concern and developing analytical tools and control methods to mitigate the three potential energy sources which can initiate accidents within a fusion reactor system: lithium, superconducting magnets and plasma disturbance.

3) development of analysis and design techniques to assess and ensure that safety risks associated with operation of magnetic fusion facilities are acceptably low.

   The programmes in this area include development of codes which analyze the fusion reactor response to an accident initiating event, establishment of a unified risk assessment methodology for use as a design tool and as a safety analysis technique and establishment of a central computerized fusion safety data base in order to have this information made easily available and accessible to designers.

PAST IAEA ACTIVITIES IN THE AREA OF FUSION SAFETY

A brief overview

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Particular interest in the subject of fusion safety was first shown by the Agency some 10 years ago, at which time the organization established the International Fusion Research Council (IFRC). At its first meeting in June 1971 IFRC recommended, inter alia, that the Agency should co-ordinate information relevant to the development of fusion power, including such topics as fusion reactor hazards. As a follow-up to this recommendation, a Consultants' Meeting on Fusion Power and the Environment was convened in December 1971. While the consultants agreed that the assessment of possible environmental effects of fusion power reactors was a very important task, they stated that estimates on tritium release from fusion reactors indicated that requirements of the International Commission on Radiological Protection (ICRP) would be satisfied. It was also concluded, however, that it was not possible at that time to proceed with further evaluation of any hazards associated with fusion reactors. The consultants recommended that further engineering studies of fusion reactors should be undertaken prior to any detailed consideration of the environmental effects of fusion power. To encourage and accelerate such studies it was proposed that a working group of experts who had made significant contributions to fusion reactor design should meet. As a result, the Agency organized the Workshop on Fusion Reactor Design Problems at Culham, United Kingdom, early in 1974, which was attended by 50 participants. Several discussion groups were set up at this workshop, among them one on "Fusion and the Environment". A summary of the discussions of this group was compiled by J.T.D. Mitchell and published in the proceedings of that workshop [1].
In the meantime, the subject of fusion power and the environment was again discussed at the second and third meetings of IFRC, in 1972 and 1973 respectively, and a request was made to the Agency by IFRC for the preparation of a detailed review on the subject. This review was published in September 1975 [2].

The subject of fusion power and the environment continued to appear as an item on the agenda of all subsequent IFRC meetings. Although it was believed by IFRC members that an early revision of the 1975 report [2] should not be undertaken, IFRC recommended that a consultants' meeting should be convened with the aim of identifying areas where further study and research work on radiological safety and environmental impact aspects of fusion were needed. This consultants' meeting was held in November 1979. The consultants prepared a report [3] and made a number of recommendations, giving the highest priority to the holding of a workshop on fusion safety in 1981. IFRC endorsed this recommendation and expressed the desire to co-ordinate this work with the work of the International TOKAMAK Reactor (INTOR) Workshop. The present workshop is the first major Agency meeting organized specifically to address the issue of fusion safety. As the last day of this fusion safety workshop is devoted to INTOR safety considerations, the INTOR Workshop will start the following week (from 30 March to 10 April 1981) and some experts will participate in both workshops, the desired degree of co-ordination will be achieved.

A few words should be said about INOTR: Work on INTOR commenced in early 1979. INTOR's Zero Phase discussions were completed about one year ago, and the Zero Phase report has been published. Currently INTOR is in the middle of Phase One. One chapter of INTOR's Zero

Phase Report is devoted to safety and environment [4], and this subject is also receiving continuing attention in the discussions of INTOR Phase One [5]. One important aspect in this regard is that, the INTOR facility should be designed such that it can be sited in any of the Agency's Member States.

The activities described above refer to the Agency's specific work in the area of fusion safety. In addition, there have been and are a number of Agency activities in the radiological safety and environmental protection areas that are not aimed specifically at fusion safety, but which are certainly relevant to fusion safety. While it would be inappropriate in the present context to list all these activities, attention must be drawn to a few important documents that have been issued. In 1978, the Agency published a report entitled "Principles for Establishing Limits for the Release of Radioactive Material into the Environment" [6]. This report presents basic concepts and principles relevant to the establishment of limits for planned releases of radioactive material from nuclear facilities and is intended for use by national authorities. The preparation of complementary documentation on the application of these principles in various practical situations is foreseen. In 1979, the Agency published the proceedings of a symposium on the "Behaviour of Tritium in the Environment"[7]. The proceedings also contain two papers that specifically address the subject of fusion [8,9]. Further papers dealing with tritium hazards are found in proceedings of other Agency meetings [10,11]. Also of interest is a recent Agency technical report on "Tritium in Some Typical Ecosystems" [12]. This publication presents data obtained in an Agency co-ordinated research programme on the environmental behaviour of tritium, which involved eleven laboratories over a period of five years. It is thought that the report will assist in predicting the behaviour of tritium in the major terrestrial ecosystems of the world. Whereas the emphasis in the report is on tritiated water, the need for further tritium research is mentioned. A comprehensive bibliography is also given. Finally, it should be noted that the Agency's Basic Safety Standards for Radiation Protection were recently revised [13]. This revision takes full account of ICRP recommendations made in 1977.

In all the activities described herein the objective is to ensure that nuclear operations are carried out safely. The workshop on fusion safety can also contribute significantly to this end.

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TRITIUM HANDLING REQUIREMENTS AND DEVELOPMENT FOR FUSION

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Abstract

Tritium technology development for the Department of Energy fusion program is taking place principally at three laboratories; Mound Facility, Argonne National Laboratory, and Los Alamos National Laboratory. This paper reviews major aspects of these three programs and discusses aspects of the tritium technology development at other laboratories in the U. S. Facilities and experiments discussed include the Tritium Effluent Control Laboratory and the Tritium Storage and Delivery System for the Tokamak Fusion Test Reactor at Mound Facility, the Lithium Processing Test Loop and the solid breeder blanket studies at Argonne, and the Tritium Systems Test Assembly at Los Alamos. Work at the Lawrence Livermore National Laboratory and the Oak Ridge National will also be discussed.

I. INTRODUCTION

Three areas of fusion technology deal with tritium handling and containment. These are the main fuel cycles including reprocessing of the reactor exhaust gas, tritium breeding and extraction from blanket systems, and tritium safety. Each area is being investigated under the sponsorship of the Office of Fusion Energy. We anticipate that early-generation fusion reactors will use the deuterium-tritium fuel cycle. The tritium fuel component is a hydrogen isotope of mass three. This material is radioactive, decaying by emission of a beta particle of 5.6-keV average energy and a half-life of 12.3 years. Tritium chemically behaves like ordinary hydrogen; reacts with oxygen to form water, with nitrogen to form ammonia and it will exchange with the hydrogen in hydrocarbons to form tritiated hydrocarbons. Because of its relatively short half-life, tritium is not an abundant isotope. Tritium is produced in nature by cosmic-ray bombardment of the earth's upper atmosphere. Neutrons, protons, and deuterons cause tritium producing reactions such as

$$n + {}^{14}N \rightarrow T + {}^{12}C$$

$$H + {}^{14}N \rightarrow T + \text{fragments}$$

$$D + D \rightarrow T + H$$

Tritiated water is formed in the atmosphere by tritium oxidation or tritium exchange with hydrogen in water through a variety of complex reactions that can be summarized as

$$T_2 + 1/2\ O_2 \rightarrow T_2O$$

and

$$T_2 + H_2O \rightarrow HTO + HT$$

Tritium normally is produced in nuclear reactors by the thermal neutron irradiation of ^{6}Li, where the following reaction occurs,

$$^6Li + n_{th} \rightarrow {}^4He + T.$$

Tritium also is produced by the fast neutron

$$^7Li + n \rightarrow {}^4He + T + n'$$

These reactions will be used to breed tritium in a fusion reactor blanket. This blanket might be a liquid lithium-metal blanket with an adjusted ^{6}Li:^{7}Li ratio, or it might be a solid lithium material such as lithium oxide, or a lithium-lead or a lithium-aluminum alloy where the thermonuclear neutrons would be captured by the lithium, thus breeding tritium.

The primary fusion reaction in a D-T plasma is the T(D,n)^{4}He reaction with an energy release of approximately 17.5 MeV, a neutron kinetic energy of 14 MeV, and a ^{4}He energy of 3.5 MeV. To obtain useful power from the energetic neutron, the reactor designers must provide a surrounding blanket in which the neutron energy is moderated by multiple collisions with the blanket materials and eventually is captured by one of the blanket or shielding components. In addition, new tritium fuel must be produced by a nuclear reaction to supply fuel to the plasma. Both objectives are accomplished by the use of lithium in the blanket. Because natural lithium is composed of two isotopes (7.42% ^{6}Li and 92.58% ^{7}Li), two nuclear reactions occur. At thermal neutron energies, the reaction ^{6}Li(n,α)^{3}H predominates with an exothermic energy release of 4.79 MeV. At higher neutron energies, the reaction ^{7}Li(n, n'α) ^{3}H increases in importance but has an endothermic Q-value of -2.47 MeV. The thermal energy from the nuclear reactions in the blanket and from the alpha particle heating of the plasma eventually will be converted to electrical power with an efficiency of ~30%; therefore, large quantities (~ 1 kg/day) must be supplied to the reactor. The quantity of fuel that must be supplied to the plasma is much larger than the amount consumed by nuclear fusion, because only a fraction of the fuel reacts during its residence time in the plasma. Based on present forecasts of particle confinement time, design studies on magnetically confined reactors project from <1% to 10% "burn-fraction" for the fuel. Therefore, it is necessary to provide rapid fuel reprocessing and recirculation of the unburned fuel for reinjection into the plasma.

Fuel recirculation requires a series of chemical and physical processing steps including evacuation of the plasma chamber, removal of impurities, adjustment of the deuterium-tritium ratio, removal of protium (^{1}H) by a hydrogen isotope separation technique, and storage of the separated fuel components before reinjection of the fuel to the plasma.

All fusion reactor designs propose the generation of new tritium fuel by the nuclear reaction of the neutrons originating in the plasma with the lithium-bearing materials in the blanket. Suggested lithium-bearing materials are liquid metal, metal alloys, fused salts, and ceramic compounds. Each material retains some equilibrium tritium level, below which tritium cannot be extracted. This equilibrium blanket level contributes significantly to the total tritium inventory in a fusion plant.

Liquid lithium has been proposed as both the breeder material and the heat transfer fluid; however, it does have a high affinity for tritium. It also poses several safety problems because of the highly reactive chemical nature of lithium. Lithium ceramic compounds have been proposed as breeder materials; here the tritium solubility in the breeder material is fairly low at the high operating temperatures of the blanket. It is proposed that this tritium would continuously diffuse from the ceramic and be removed in the stream of an inert gas. Tritium diffusion, either as the chemical species T_2 or as T_2O, through the ceramic is relatively slow; consequently, the particle size must be kept small to prevent excessive tritium "hold-up."

A D-T burning fusion machine will require large tritium inventories. The Tokamak Fusion Test Reactor (TFTR) at the Princeton Plasma Physics Laboratory will have a tritium inventory of 5 g or less, and there will be no on-site fuel reprocessing. However, the tritium inventory at a next generation machine (perhaps the Fusion Engineering Device, the International Tokamak Reactor, or the Engineering Test Facility) will require a few kilograms. This inventory, in itself, will be a large source of radioactivity (1 g tritium $\simeq 10^4$ Ci) at a fusion site. Adequate personnel and environmental protection and safety systems must be integral parts of the tritium facility at such a fusion machine. The development and evaluation of these safety systems must be accomplished in a timely manner so that these issues do not pose a threat to public acceptance of fusion as an energy source.

Other special problems introduced by tritium are related to radiation damage and chemical compatibility of materials used in tritium facilities. The tritium itself is a radiation source, although the tritium beta-particle is a low energy source. This beta-particle can cause radiation degradation in some materials, notably in the hydrocarbons, including elastomers that might be used for gaskets, valve seats, etc. A special problem exists with the use of perfluorohydrocarbons, such as Teflon and viton, because the tritium undergoes exchange reactions with the fluorine:

$$T_2 + (-CF_2-CF_2-)_n \rightarrow TF + (-CTF-CF_2)n$$

This reaction leaves an internal radiation source as a component of the elastomer; however, a more serious problem is generation of the extremely corrosive and radioactive tritium fluoride, (TF). For this reason, the use of fluorinated hydrocarbons must be avoided in the design of tritium systems. Also, because tritium will cause hydrogen embrittlement, as do ordinary hydrogen and deuterium, materials used in a tritium facility must always be based on the use of materials which are known to be resistant to hydrogen embrittlement.

II. CURRENT ACTIVITIES IN TRITIUM TECHNOLOGY

A. Los Alamos National Laboratory

In 1977 the Office of Fusion Energy, US Department of Energy, funded the Los Alamos National Laboratory to design, construct, and operate the Tritium Systems Test Assembly (TSTA)[1]. The principal objectives of the TSTA Project are to: (1) demonstrate the fuel cycle for fusion power systems; (2) develop, test, and qualify equipment for tritium service in the fusion program; (3) develop and evaluate personnel and environmental protection systems; (4) provide a facility that will yield a reliable data base for tritium handling systems for future fusion facilities; (5) demonstrate long-term safe handling of tritium with no major releases or incidents; (6) investigate and evaluate the response of the fuel cycle and environmental packages to normal, off-normal, and emergency situations; and (7) develop tritium-compatible components with long-term reliability.

The TSTA schedule calls for the facility to be operational by the end of 1981. Modifications to an existing building at Los Alamos have been completed (Fig. 1), and installation of equipment for specific subsystems now is under way. The project is more than halfway through the design-construction phase and currently is on schedule for the 1981 operational milestone. The goal of the TSTA project is to provide an extensive data base for the designers of the first large-scale, D-T burning fusion machine, probably the Fusion Engineering Device (FED) or the International Tokamak Reactor (INTOR). The FED concept is being developed through the Engineering Test Facility Design Center at Oak Ridge National Laboratory. Optimistic estimates indicate that a FED could be operational during 1991-1992 but would require a detailed design phase during 1983-1987. To make a significant contribution to the FED data base, TSTA must become operational by early 1982. The INTOR project is a joint international effort involving the United States, Japan, the USSR, and the European Community, and its goals and objectives are quite similar to those of FED. At this time, the four INTOR participants are engaged in workshop and conceptual design meetings. Realistically, we cannot predict that INTOR can be built any faster than FED can. Therefore, the current TSTA schedule will also provide the necessary data base for INTOR.

The TSTA will consist of a large interactive gas loop (Fig. 2) that simulates the proposed fuel cycle for a fusion facility. The reactor torus will be simulated by a vacuum vessel into which gas mixtures are introduced at the compositions and pressures predicted for an actual reactor torus at the end of a burn cycle. This gas mixture, primarily $(D,T)_2$ containing a variety of impurities, must be evacuated through a prototypical vacuum system; impurities must be removed from the $(D,T)_2$; and isotopic separation must be performed to produce D_2, T_2 and DT. The resulting gases then will become the fuel components that are injected into the reactor (vacuum vessel) for the next burn cycle. The gas loop is designed to handle up to 360 g-mol/day (1800 g) of D-T. This flow will provide cycle-operating experience on a scale similar to that being considered for FED and INTOR. In addition to the gas loop, all safety and experimental systems associated with such an extensive tritium facility are under consideration. To accomplish the program goals, an on-site tritium inventory of approximately 150 g is required. This report discusses the current status and design of each major TSTA subsystem.

1. Vacuum Facility (VAC). This system consists of a large vacuum vessel with associated duct that leads to the vacuum pumps being evaluated for use in the fusion program. Three cryo-based vacuum pumps, candidates for the primary vacuum pumps on a fusion reactor, are being evaluated at TSTA. These are "compound" pumps because they are two stage. The first stage is cryocondensation of hydrogen on a metal surface cooled to or near liquid helium temperatures. This stage will pump all the hydrogen isotopes, but not the helium isotopes. In the first pump (Fig. 3) built by Los Alamos National Laboratory [2] , the helium is pumped by cryosorption on a molecular sieve surface cooled to liquid helium temperatures. The second pump, developed at Brookhaven National Laboratory [3] , will pump helium by cryosorption on a charcoal surface cooled to liquid helium temperatures. The third pump, built by the Lawrence Livermore National Laboratory (LLNL), will pump helium by argon cryotrapping. A pump very similar to that being built for TSTA by LLNL has been described by Batzer et al. [4]. In this pump a fine spray of argon gas will be injected into the pumping area and will condense as a solid on a helium cooled metal surface. The helium gas will be trapped and thus pumped by the argon ice formed on the cold surface. All three pumps have been built and are undergoing performance testing. The Los Alamos built pump has demonstrated that a compound pump can simultaneously pump helium and hydrogen isotopes. The separation of hydrogen, on the condensation panel from helium on the cryosorption panel, which is extremely sharp, can be maintained by careful temperature and pressure control during regeneration. The helium panel first is regenerated, then the cryocondensation panel can be warmed to remove hydrogen from the pump. Complete evaluation of all three pumps will continue.

The TSTA vacuum facility also will include a complete regeneration system to be used during regeneration of the cryopump. The regeneration system has been described by Coffin and Walthers [2].

2. Fuel Cleanup (FCU). The primary function of the FCU is to separate all of the other atoms present from the hydrogen isotopes in the gas stream from the VAC facility. This step is essential. If other molecular species, such as tritiated water, methane or ammonia, and argon, were present in the hydrogen isotope stream, they would freeze and plug the low temperature distillation columns of the Isotope Separation System. The FCU not only must separate $(D,T)_2$ and (H,D,T) from the reactor of offgas contaminants, it also must recover as $(H,D,T)_2$ all hydrogen isotopes that are chemically combined with other atoms in the reactor offgas, i.e., $C(H,D,T)_4$, $N(H,D,T)_3$ and $(H,D,T)_2O$. Table I shows the predicted feed stream flows to the FCU. The hydrogen molecules can be separated from the other molecular species by adsorbing the impurities at 75 K on a 5-A molecular sieve. (The tritiated ammonia and water are removed from the offgas before it enters the molecular sieve.) Regenerating the molecular sieve requires additional processing. These considerations have led to two proposed processing schemes (Fig. 4). In one system, a front-end, hot-metal-bed (uranium at 1170 K) removes carbon, nitrogen, and oxygen from the feed stream by converting them to uranium carbides, nitrides, and oxides and releasing the associated hydrogen isotopes as gas. Periodically, the uranium will become saturated with impurities and must be replaced. The second front-end system has a catalytic reactor to convert any free oxygen in the feed stream to $(D,T)_2O$, which is then removed, along with the ammonias and carbon dioxide by freezing. These two front-end packages would be operated alternately. Each front-end system is backed by a cryogenic package where argon and any other trace impurities are removed by adsorption on molecular sieve, thus producing a pure stream of hydrogen isotopes to feed the isotope separation system. The FCU has been described in detail in a previous paper [5].

Table I
Impurity Gas Load for TSTA
(based on 360 g moles/day DT flow)

Element	Mol%	Species	Grams/day
He	2-20	He	29-290
H	1	HD, HT, H_2	7
C	0.02-0.01	$C(D,T)_4$, $C_2(D,T)_2$ CO_2	1.7-9
N	0.01	$N(D,T)_3$, N_2, CN, NO_x	10
O	0.5	$(D,T)_2O$, CO_2, O_2	57
Ar	0.00006-0.05	Ar	0.17-14

3. Isotope Separation System (ISS). At TSTA, cryogenic fractional distillation [6,7] is being used for hydrogen isotope separation. A system that uses four interlinked columns, with chemical equilibrators located between columns 1 and 2 and between columns 2 and 4 (Fig. 5) has been designed. The system is sized to handle the full flow appropriate to ETF or INTOR, i.e., 360 mol DT/day. It also will handle the simulated flow from the neutral beam channel vacuum pumps (~275 mol D_2/day). Figure 5 shows the flow paths and purities of the major components in each of the four output streams. Refrigeration will be supplied by a central cryogenic refrigerator that provides helium gas at 20 K. In case of refrigeration loss, the ISS contains two systems for storing gas evolved from vaporization of the hydrogen liquids.

First, a large surge tank allows the total contents of the ISS to be stored, as a gas, at a pressure <100 psi. Second, each column has access to a vessel filled with uranium powder that can hydride the gaseous hydrogen isotopes and thus store them as solids. The contents of each column can be stored individually during periods of planned shutdown. The ISS is totally under double containment. The distillation system now is installed at TSTA. Performance tests by the manufacturers before shipment of the columns to the Laboratory demonstrated an excellent separation of H_2, HD, and D_2 in the columns. Tests with tritium will not, of course, be performed until tritium operation of TSTA occurs.

4. Transfer Pumps (TPU). The transfer pumps at TSTA will provide circulation and transport for mixtures of hydrogen and helium isotopes from one portion of the flow loop to another. An early decision was that

all process-wetted components of transfer pumps would be of metal or inert carbon construction. Mechanical motion of internal parts is transmitted by flexible metal membranes (diaphragm or bellows), by magnetic transmission (canned motors), or by magnetically confined sealants (Ferrofluids). Most TSTA gas transfer requirements are met by a metal-bellows pump (Model MB-601, manufactured by Metal Bellows Corp., Sharon, Massachusetts) that incorporates a replaceable, all-metal, check valve assembly designed and tested at Los Alamos. The basic pump has two heads, which can be plumbed to operate in series, in parallel, or independently. Figure 6 shows our concept of the TSTA metal-bellows transfer pump housed in the secondary containment glove box. Other pumps being considered include an all-metal, bellows-sealed, stainless steel scroll pump and a canned-motor, Ferrofluid-sealed Roots blower. These pumps are being tested and installed at TSTA.

5. Emergency Tritium Cleanup (ETC). This system will process all of the air in TSTA if a gaseous tritium release to the facility occurs [8]. It will recover most of the released tritium, thus reducing losses and environmental impact. The TSTA cell contains 3000 m^3 of building atmosphere that would become contaminated with tritium if an accident caused the primary and secondary containments to be breached. The flow rate through the ETC is 0.65 m^3/s. The ETC will be an automatically actuated, room-air detritiation system based on a precious metal catalytic recombiner where hydrogen isotopes are oxidized to water. The water is collected, partly as liquid water and by adsorption on molecular sieve beds. The flow path for gas through the ETC is shown in Fig. 7. A 100-g T_2 spill into the facility would give an initial concentration of ~355 Ci/m^3 in the cell, and the ETC would reduce the room level to 40×10^{-6} Ci/m^3 within 24 h. The ETC at TSTA was designed with individually purchased components, and the assembly and installation of these components, including instrumentation, is being done by Los Alamos. The ETC equipment has been bought, and the installation of this system is nearing completion.

6. Tritium Waste Treatment (TWT). This system provides routine processing of all gaseous effluents generated at TSTA to remove tritium from these effluents before they are released to the environment. It is based on the design of a similar system that has been operating for 5 yr [9] and that has been discussed recently by Nasise [10]. The TWT is designed to operate at a flow rate of 15 or 60 scfm, depending on the gas load at the TWT inlet.

The TWT is a computer-actuated and -controlled tritium removal system that operates by the catalytic conversion of all hydrogen isotopes in the input stream to water and organic materials oxidized to water and carbon dioxide. The oxygen level will automatically be maintained in the system at levels sufficient to ensure catalytic conversion of all hydrogen isotopes to water. The water generated by these processes will be adsorbed on a molecular sieve. Then, the remaining gaseous effluent will be discharged to the atmosphere through the building ventilation system after the tritium has been removed to an as-low-as-practicable level. Figure 8 is a flow chart of the TWT. As with the ETC, the TSTA staff designed the TWT and purchased individual components for assembly in the final system, rather than buying a preassembled TWT package from a single supplier. All components for the TWT are now installed at TSTA. After final piping and electrical connections are completed, the system will be ready for testing.

7. Tritium Monitoring (TM). Tritium monitoring instrumentation will perform several key functions, including quantitative determination of stack releases, assurance of personnel safety, initiation of cleanup of experimental rooms and secondary containment volumes following tritium releases, monitoring and controlling the operation of the main process loop and the cleanup systems, and monitoring the performance and results of the experimental contamination studies program. For the most part, monitors will be flow-through ionization chambers with redundancy provided for critical situations. The monitors will, for the most part, be similar to currently available standard tritium instrumentation.

Major differences may include special sensors (e.g., plastic scintillators) required for line monitors and alpha-rejection circuitry to increase sensitivity and selectivity. All of the monitoring equipment for TSTA has now been defined [11] and is being procured and installed at this time.

8. Secondary Containment (SEC). The philosophy at TSTA is to doubly contain all components of the primary fuel process loop wherever tritium conceivably could be released in multicurie quantities, posing significant hazards to workers and the environment. Secondary containment concepts being applied to TSTA include double-wall piping and components, gloveboxes or other large volumes housing tritium-wetted components, and integral vacuum jackets around cryogenic components. Vacuum jackets help provide thermal isolation as well as secondary containment. Initial plans did not call for secondary containment of the VAC facility because we felt that the primary containers (vacuum tank and cryogenic pumps) could be designed with a safety factor that would virtually eliminate the risk of a release. This decision was reversed and secondary containment of VAC will be provided. In all other subsystems of the main fuel process loop, the secondary containment is being designed and installed with the subsystem.

9. Experimental Contamination Studies (XCS). A small laboratory at TSTA will be dedicated to the study of several aspects of tritium contamination/decontamination. We included plans to study the performance and efficiency of a small tritium-cleanup, gas-detritiation system; surface contamination, outgassing, and permeation properties of construction materials (wood, concrete, steel, etc.) and surface coatings (epoxy paint, latex paint, etc.); contamination and outgassing properties of metals, glasses, and plastics used in tritium-containment systems; the relationship between total tritium in a material and the amount measured by surface-survey probe and by wiping removable tritium; methods of decontamination; and gas conversion rates of T_2 and D-T to the oxide forms during realistic release situations. The XCS will be housed at TSTA in a separate laboratory room that can be completely

sealed off from the rest of the facility. Then, evaluation of the ETC can be made by releasing tritium into the sealed room. Under these conditions, the ETC, operating at 25% of normal speed, will be used to detritiate the XCS room. When the XCS experiments have generated sufficient data on surface coatings for tritium facilities, the XCS room will be refinished to provide the best surface coatings for walls, ceilings, and floors. The tritium release/cleanup experiments will be repeated. These experiments will be very valuable in choosing construction materials, surface coatings, etc., for future fusion systems. The equipment for this laboratory is being designed and obtained; it will be installed and evaluated in 1981.

10. Master Data Acquisition and Control (MDAC). The TSTA is designed to be a computer controlled system and will not operate fully unless MDAC is operational. Each subsystem will be able to be tested without MDAC if necessary; however, at the present time, it is anticipated that the process loop of TSTA will not operate without MDAC control. The MDAC will incorporate features to minimize potential hazards to operational personnel, the general public, and equipment. All identified hazards will be monitored and controlled by hard-wired interlocks and backed up by the monitoring of MDAC. The MDAC will monitor all radiation detectors and take appropriate action (give alarms, advise of building evacuation,) if unsafe conditions are detected. The TSTA subsystems will be self-protecting to insure against computer error resulting in a hazardous operating mode. Some equipment that may lead to hazardous situations during malfunction will have built-in, absolute-limit protection to ensure against both local-manual and remote-computer errors. Checks of equipment performance will be done in software. The MDAC also will monitor input commands from critical locations to ensure that neither operations by unauthorized personnel nor errors by authorized operators will cause hazardous situations. Validation checks on computer commands will be performed in software. The computer, a Data General Eclipse, has been purchased and is installed and operating at TSTA; control software is being developed. The MDAC

will use a CAMAC interface system. The computer system is operated by an Uninterruptable Power Supply, but an Emergency Generator Set will be available to operate the MDAC and critical components of the major subsystems if a commercial power loss should occur.

B. Argonne National Laboratory (ANL)

The fusion reactor research program at Argonne National Laboratory includes several applied research topics covering a variety of interrelated areas of fusion tritium technology. This work focuses principally on studies of fuel handling, breeder blanket processing, and tritium containment. These studies are both experimental and analytical in character and, in recent years, have spearheaded technological advances many important fusion-specific areas which are described below.

Modest advances have been made in recent years in the study of liquid lithium processing. A 200-liter-capacity system, the Lithium Processing Test Loop (LPTL) [12] has been operating for over 9500 h. (Fig. 9) Cold trapping, reactive-metal getter trapping, and a method based on molten salt extraction as a technique for removing trace impurities from liquid lithium have been tested [12] using the LPTL and related facilities. The ranges of the projected lower-limit impurity control levels for the elements H(D,T), O, N, and C, based on these tests, are shown in Fig. 10. Molten salt extraction offers the best method for developing a regenerable process to recover tritium (from D-T reactor blankets) and to control impurities at the <10 appm level. The salt extraction method also seems applicable to the processing of liquid lithium-lead alloys.

In the area of instrumentation and hardware for liquid lithium systems, permeation- and resistivity-type meters have been developed that can be used effectively for monitoring hydrogen [13] and hydrogen plus nitrogen [13]. The principal difficulties in operating present stainless steel lithium systems have resulted from (1) cracking of special components containing cold-worked material with high residual stress (EM-pump channels and valve bellows)[12] and (2) mass transfer buildup [12,14] involving iron, chromium, and nickel. A method developed by DeVan et al. [14], wherein aluminum is added to lithium to react with and stabilize the surfaces of stainless steel components, has shown promise as a means of retarding mass transfer and intergranular penetration in stainless steel systems.

The use of solid lithium compounds as the breeding material for D-T fusion reactors has been studied increasingly in the USA in recent years [15]. Three different approaches for tritium extraction have been suggested: (1) in situ tritium recovery, (2) removal and external processing of fuel-pin breeder assemblies on a periodic basis (every 6 to 12 months), and (3) continuous circulation of solid material into and out of the reactor with tritium processing done externally. Evidence suggests that the in situ recovery of tritium is feasible, but a carefully controlled breeder material temperature distribution is required [15] so that tritium does not build up to excessive levels in low-temperature regions and sintering does not occur in high-temperature regions.

The fuel-pin approach to tritium breeding has the problem of large in-blanket tritium inventories (~50 kg/fusion GW for annual pin removal), whereas the circulating-solid approach presents formidable engineering complexities. However, if the in situ method of handling solid breeder blankets becomes intractible and if liquid metals are eliminated from consideration because of engineering or safety considerations, the fuel-pin and circulating-solid approaches may prove to be the only recourses for self-sustaining D-T fusion reactors.

In work related to the STARFIRE Commercial Fusion Reactor Study [16], at Argonne, a top-down selectivity analysis has been made [17] to identify the most tractible breeder/coolant/structure (B/C/S) combinations for D-T fusion reactors. In a typical analysis, a breeder material (liquid lithium, liquid lead-lithium alloy, solid Li_7Pb_2, Li_2O, or another lithium-containing ceramic) is matched with various coolants (water, helium, liquid metals, molten salts), and compatibility assessments are performed. The criteria for these assessments are breeding capability, exothermicity of breeder/coolant interaction, operating temperature requirements, and thermodynamic efficiency of the associated power cycle. Breeder/coolant combinations that survive the scrutiny of this first assessment are matched with likely classes of structural materials (austenitic-, nickel-, ferritic-, and refractory-base alloys) and a second assessment is made. For the second assessment, criteria are based mainly on structural material compatibility (with breeder and coolant) and temperature constraints. In summary, no B/C/S combination from the choices listed above has been developed to the point where we can confidently predict that satisfactory performance in the fusion environment can be achieved. Liquid lithium/ refractory metal (Nb- or V-base alloys) ceramic breeder/water (pressurized or boiling)/ austenitic or /ferritic combinations appear to be among the least objectionable choices from an engineering and compatibility point of view.

C. Mound Facility (MF)

The Mound Facility has been actively involved in tritium technology for over 20 years. Recently, Mound has focused most of its tritium technology development on tritium containment and environmental control. The two components of Mound's tritium technology development currently most active and most relevant to fusion needs are the Tritium Effluent Control Laboratory (TECL) and the Tritium Storage and Delivery System (TSDS) project for the Tokomak Fusion Test Reactor (TFTR) at Princeton.

The TECL was initiated in 1971 for development and demonstration of tritium containment. The goal of TECL is to prevent any tritium release to the environment and to recover for reuse all tritium released within the laboratory. The TECL consists of an integrated set of containment systems and detritiation experiments. Tritium containment is provided by gloveboxes, a glovebox atmosphere detritiation system (GADS), a laboratory that can be isolated from the remainder of the building, and an emergency containment system (ECS). The GADS is a 0.05 m^3/s helium purifier that continuously cleans the glovebox atmosphere. The ECS is a 0.5 m^3/s catalytic oxidation, water-vapor-adsorption system for room air detritiation. In addition, a 7.5×10^{-3} m^3/s utility air detritiation system of the oxidation-absorption type is available for cleaning passbox atmospheres and for nonstandard applications [18] .

Four tests have measured the efficiency of the GADS and ECS in cleaning up after a tritium release in a glovebox or a room. Glovebox cleanup was as predicted; however, ECS room tests showed substantial tritium loss through imperfect exhaust duct seals. More tests are planned to study surface absorption and conversion to HTO. This was the first test of such a large ECS system in the U. S.

Tritiated water vapor collected by the ECS or by similar systems can be detritiated by the combined electrolysis catalytic exchange (CECE) pilotscale unit that is part of TECL. The CECE incorporates a countercurrent flow of water and hydrogen gas in two 2.5-cm diam, 7.5-m-long columns packed with a hydrophobic, precious metal catalyst developed by Atomic Energy of Canada, Limited [19]. Bottom reflux is provided by a solid polymer electrolysis unit, whereas top reflux is provided by a catalytic recombiner. Tritiated water containing 300 Ci/liter has been stripped to 10^{-3} Ci/liter in the CECE at feed rates of approximately 4 ml/min of water. Although CECE has not been tested long enough to accumulate an equilibrium concentration, values in the range of 1-10 mCi/liter are expected [20]. The goal of the CECE work is to develop a full scale detritiation plant suitable for processing tritiated acqueous waste from fusion or fission reactors and from fuel reprocessing plants.

Part of the hydrogen generated in the electrolysis unit can be withdrawn to use as feed for the remaining TECL component, a cryogenic distillation system. This system includes a single 0.6-cm-diam x 50-cm-long packed column, operating at approximately 25 K. Bottom and top concentrations of 2500 Ci/m^3 and 10^{-3} Ci/m^3, respectively, were measured during one run with this column, for an enrichment factor of 2.5×10^6. Feed rates of 100 std cm^3/min are possible at this enrichment factor. This work is supported by computer simulation studies of the distillation process. In addition, an experimental study of the hydrogen isotope equilibration rate, ($2DT \rightleftarrows T_2 + D_2$) at cryogenic temperatures is being performed. This study is of interest both in cryogenic distillation and in cryogenic fuel-pellet production.

The Tritium Storage and Delivery System (TSDS) has been designed and constructed for use in the Tokamak Fusion Test Reactor (TFTR) at the Princeton Plasma Physics Laboratory. The TSDS will receive, assay, store, and deliver measured quantities of high-purity tritium to fuel the TFTR. The TSDS consists of a receiving manifold, uranium tritide storage beds, transfer pumps, and metering volumes, all contained in two stainless steel gloveboxes. In addition, a quadrupole mass analyzer and pressure-volume-temperature measuring equipment permit assay of the tritium as received, as well as confirmation that the tritium delivered to the reactor has the required purity. Extensive development was done with the mass analyzer to permit analysis of the required sensitivity and accuracy [21].

In operation, tritium received at TFTR is introduced to the TSDS through the receiving manifold; after being assayed, it is pumped onto a uranium bed where it is stored as uranium tritide for future use. During D-T operation of TFTR, an appropriate tritium storage bed is heated to raise the internal tritium pressure to about 1/2-1 atm. Each time the reactor is to be fueled (as often as every 5 min), the required amount of tritium is pumped into a metering volume. The gas then passes through a delivery manifold to three calibrated injection volumes near the torus. Finally, injection to the torus is controlled by a specially designed piezo-electric valve at each injection volume. After a burn, the fuel is not recycled but is recovered by the TFTR vacuum system and stored for later reclamation.

The TSDS is designed to achieve the highest possible levels of reliability, safety, and tritium containment. The two transfer pumps are interchangeable, two-stage, doubly contained, metal-bellows pumps, one of which is a spare. The three interchangeable storage beds are each doubly contained in stainless steel, with a provision for purging the secondary volume of tritium that permeates from the primary container. Each bed has a rupture-disc-sealed connection to an evacuated "dump" tank. At any given time, one storage bed is active, one is available for cleaning operations, and one is a spare.

Complete instrumentation is provided for TSDS to permit accurate control and to promote safety. The tritium generators are provided with pressure sensors, redundant thermal control, and over-temperature protection.

Normal operations of the TSDS are performed remotely through computer control of pneumatically operated valves. All process valves are bellows sealed with polyimide instead of metal. These valves have been remachined individually to ensure reliable leak-free operation.

The TSDS will be tested with a load of 10^4 Ci of tritium to ensure reliable operation of the entire system before delivery to the Princeton Plasma Physics Laboratory. A dummy manifold and injection chamber will simulate the manifold and torus at Princeton.

D. Lawrence Livermore National Laboratory (LLNL)

Several areas of tritium technology of interest to the fusion energy community are being pursued at the Lawrence Livermore National Laboratory. One area of concern is the tritium recovery following an environmental or atmospheric release of tritium. In current designs of systems for recovery of such released tritium, the method of recovery is catalytic oxidation with atmospheric oxygen forming tritiated water, which then is collected. Sherwood [22] has measured room temperature kinetic data on tritium/air oxidation with three common catalyst/substrate formulations. These three formulations were platinum/alumina, palladium/kaolin, and palladium/zeolite. Compared to self-catalyzed atmospheric conversion, each of the dispersed-metal catalysts is extremely effective in promoting tritium oxidation; equivalent first-order rate constants are higher by roughly nine orders of magnitude. Electron-microprobe scans reveal that the dispersed metal is deposited near the outer surface of the catalyst, with metal concentration decreasing exponentially from the pellet surface. The platinum-based catalyst is more effective than the palladium catalyst on a surface-area basis by about a factor of three.

Miller et al. [23], at LLNL, are working on the development of an organic getter which will operate in the presence of air and will minimize the formation of the more hazardous tritiated water. The compound of interest, 1,4-diphenylbutadiyne, is a hydrocarbon, $C_6H_5-C\equiv C-C\equiv C-C_6H_5$. The T_2 adds to the acetylene (triple) bond of the getter in the presence of a metal catalyst. However, the necessary catalyst will stimulate the $T_2 + O_2$ reaction, so some tritiated water will be formed. Early results indicate that these getters will indeed remove tritium from air, although with the formation of some water. The organic triple bond appears to be a very good candidate for this type getter. Miller states that the best solution would be to combine the metallic catalyst and the triple bond in a single fairly simple molecule, for instance $(PhC\equiv CPh)_2Pt$, where Ph is a phenyl group. To date, very little work has been done on methods of disposing of or storing these tritiated organic getters. This work will be done when an optimum getter material is chosen, and the gettering process is thoroughly understood.

The Rotating Target Neutron Source (RTNS-II) at LLNL [24] is an accelerator-based neutron source for studying radiation damage to materials. Energetic deuterons bombard a solid metal tritide target, producing fusion neutrons (over 10^{13} 14-MeV neutrons/sec). Deuterium continually displaces tritium from the target at rates as high as 7 Ci/h. The anticipated addition of a second accelerator and increased neutron yield of the present accelerator could increase tritium output to 20-30 Ci/h. This tritium is released into the accelerator vacuum system. Because it is not acceptable to release this tritium to the environment, a tritium-scrubbing system was devised [25] to clean the vacuum system exhaust before venting it to the atmosphere. This system consists of a catalytic recombiner, where tritiated water is formed, and molecular sieve drying towers for collecting the water. When these molecular sieve driers become saturated, they are replaced and the saturated beds are buried. The driers contain approximately 7 pounds of molecular sieve loaded to 14-15% of the dry weight. At current operating levels, water loads of ~0.25 lb/wk are collected. Most of this is D_2O, with only ~1% being tritium. Schumacher reports a 10^5-10^6 tritium concentration reduction factor through the scrubber system, with most of the escaping tritium in the gaseous form. Small scrubber systems, such as the one at RTNS-II, may find extensive use in the fusion program.

Souers [26], at LLNL, is measuring and correlating cryogenic data on D_2, T_2 and mixtures of these components. Because solid D-T may be a future fusion fuel, the measurement of physical and chemical properties of cryogenic D-T in the solid, liquid, and gas phases will aid the design engineer and the plasma physicist considering the use of cryogenic D-T.

Some of the properties being measured include the D-T reaction rate, that is, the rate at which $D_2 + T_2$ react to form the three-component mixture D_2-DT-T_2. At room temperature, this reaction takes place with an exponential 1/e-time on the order of tens of minutes. At 20 K, the 1/e-times are on the orders of tens of hours. Other properties being measured by Souers include thermal conductivity of the solid mixtures and electrical conductivity of both liquid and gaseous species in the 20-26 K range. Souers has compiled a very important report [27] in which he correlated the measured physical and chemical properties of the hydrogens <30 K. This compilation will benefit fusion engineers who do not have a strong background in cryogenic materials, as well as those specialists who will be concerned with details of the low temperature D-T mixtures.

E. Oak Ridge National Laboratory

Bell et al. [28] have measured the tritium permeability of structural materials and surface effects on permeation rates. Tritium management in any system always should include containment so that tritium release rates will be less than established limits and established limits will be as low as practicable. The ability of hydrogen isotopes to permeate most materials makes complete tritium containment an extremely difficult task. However, tritium release rates from a given system can be minimized by two primary efforts. First, the selection of a compatible containment material which, frequently, also will be the structural material. This effort would include finding films and barriers compatible with the structural material. Second, and sometimes more important, the material should have surface chemistry that continuously impedes tritium permeation. Bell has shown that by exposing the austenitic Incoloy 800 and the ferritic SS 406 to steam oxidation at 0.94 atm and 930 K, the permeabilities of these materials can be reduced by several orders of magnitude over the permeability of the clean metal (Fig. 11). These studies demonstrate that in situ surface oxidation of construction alloys can produce oxide barriers that reduce tritium permeation by significant factors. However, these results have limitations, when applied by extrapolation, to operating systems with high temperatures and corrosive conditions. Therefore, Bell emphasizes [28] that current conclusions of permeation barrier effects must be considered only as indications of effects under actual operating conditions. This type of experiment needs to be repeated on a larger scale where conditions more nearly match reactor type conditions. The demonstration that, indeed, these permeation barriers can be formed in situ, under reactor conditions, would be a very important development for the fusion energy program.

III. CONCLUSIONS

Although this discussion shows the breadth of tritium technology development in the U. S., it cannot begin to cover all the many research and development programs in the field. We hope to show that there is a large, serious effort underway to develop this technology in a timely manner. In the current programs, tritium technology should not be an obstacle to the design and construction of the Fusion Engineering Device (FED). These continuing programs will develop the data base and experience necessary to proceed from conceptual design to engineering design and construction of the FED.

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FIG. 1. The TSTA building.

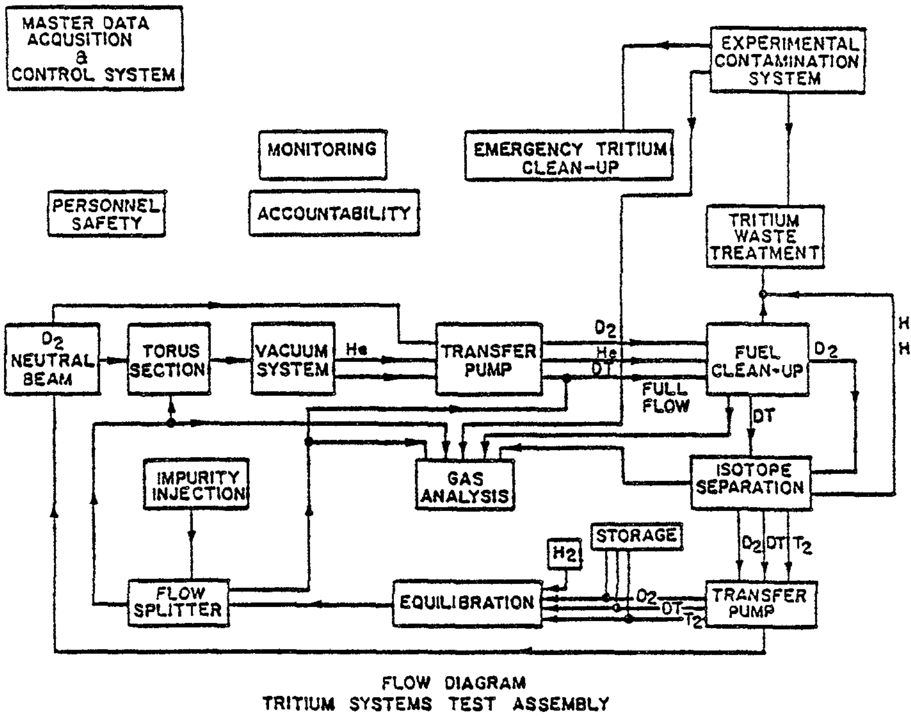

FIG. 2. The TSTA process loop showing subsystem interactions.

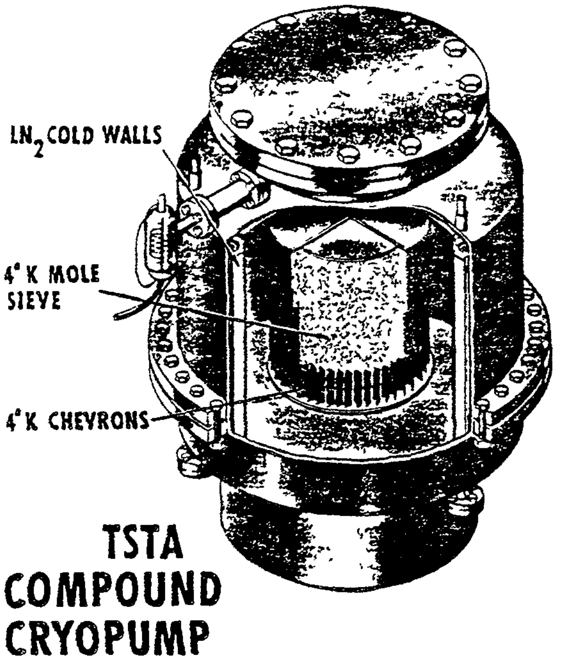

FIG. 3. The Los Alamos TSTA compound cryopump.

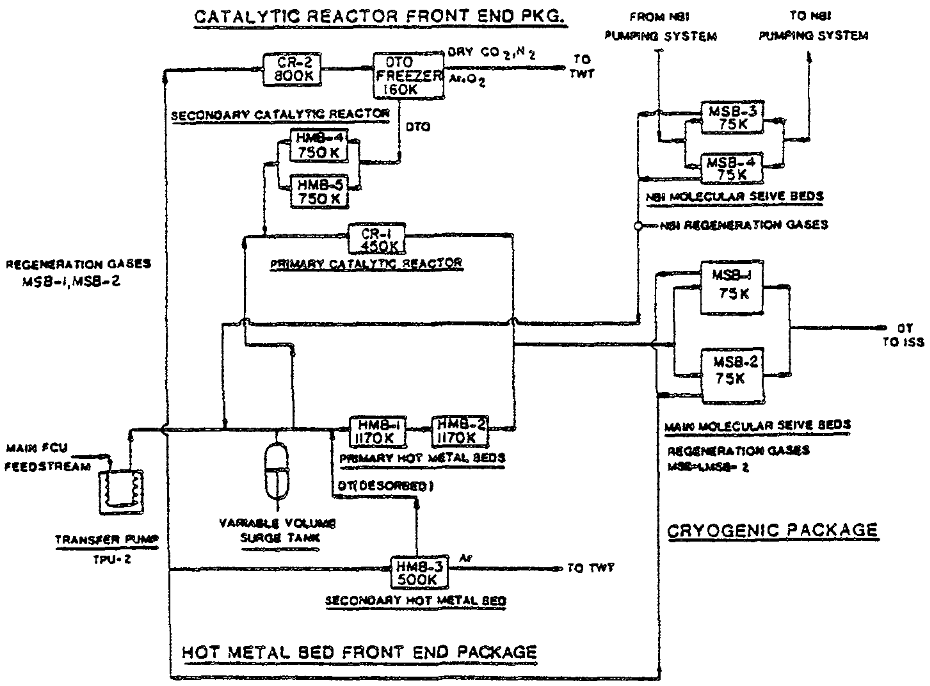

FIG. 4. TSTA fuel cleanup flow schematic.

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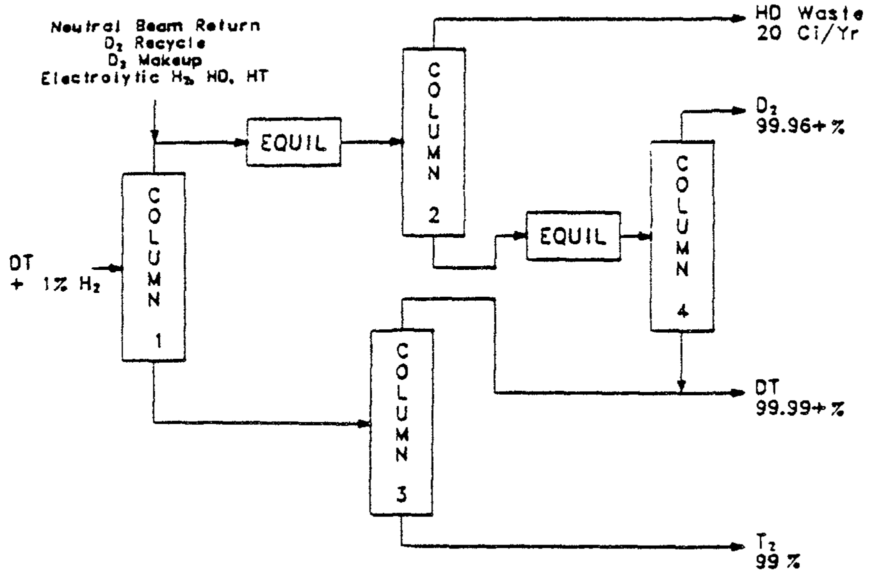

FIG. 5. Flow diagram for the ISS distillation columns.

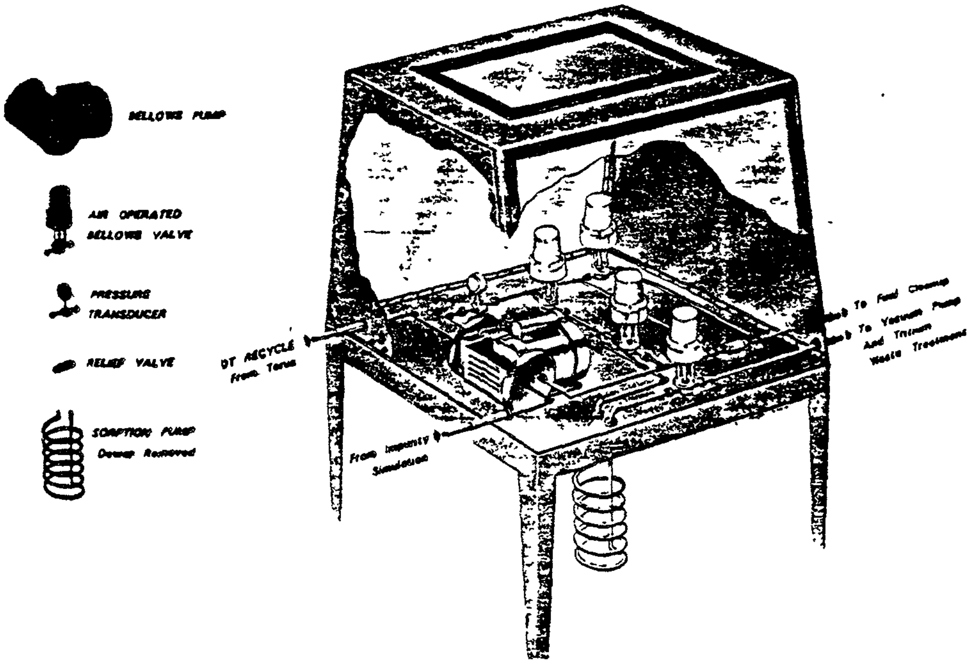

FIG. 6. Transfer pump unit enclosed in secondary containment.

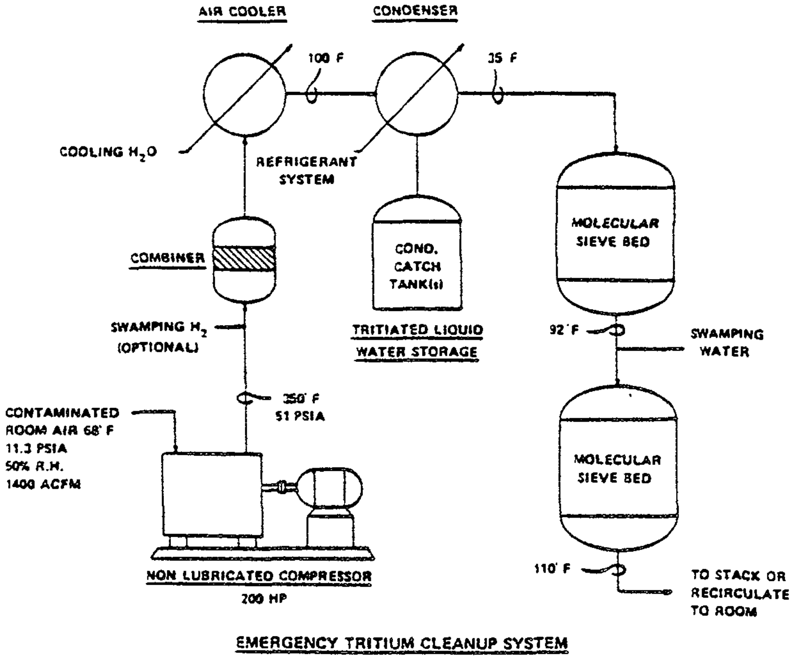

FIG. 7. The Emergency Tritium Cleanup System (ETC).

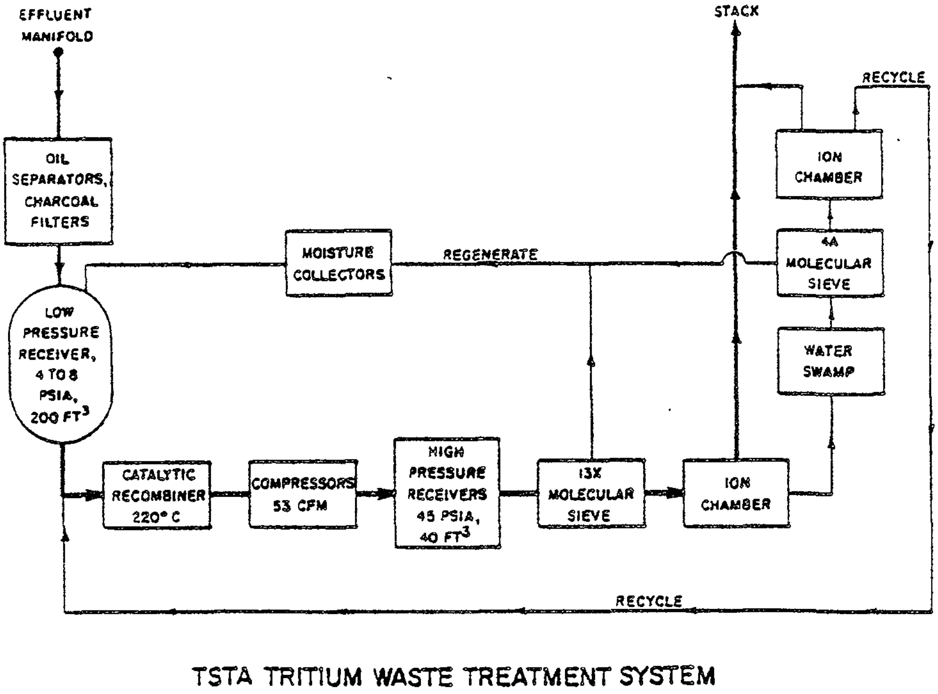

FIG. 8. The Tritium Waste Treatment System (TWT).

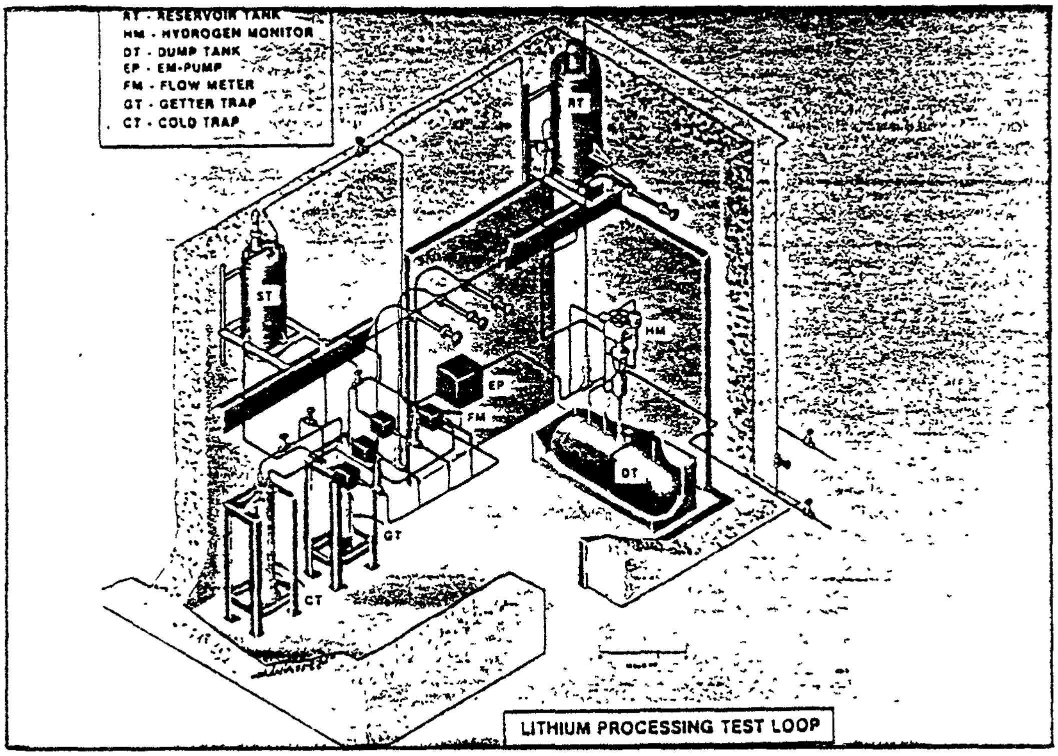

FIG. 9. The ANL Lithium Processing Test Loop (LPTL).

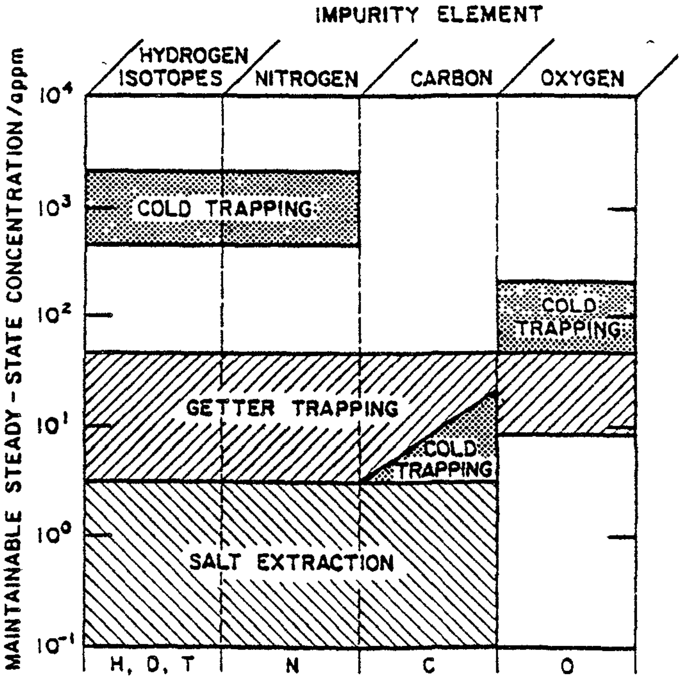

FIG. 10. Summary of projection of achievable lower-limit control levels for selected lithium processing methods.

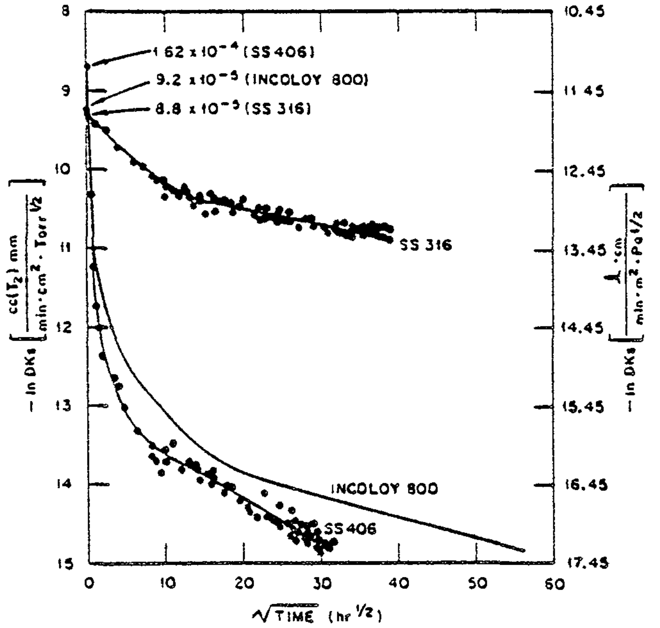

FIG. 11. Tritium permeabilities of three construction alloys at 930K as the alloys are oxidized by water at 0.94 atmosphere.

DISCUSSION

J.G. CROCKER: It might be interesting to you to know that the TFTR people do have three persons at the Mound Laboratory being trained on the tritium equipments being built there to make sure that those who have not really had a lot of experience in tritium handling before know what to do when the tritium will be used.

J.GRUBER: What T/H ratios will be considered as waste ?

J.L. ANDERSON: We anticipate that in the exhaust gas from a reactor there may be as much as 1% hydrogen impurity, protium in the deuterium and tritium. That 1% is not a significant fraction but it is a fraction that has to be removed. We feel that in our isotopes separation system we can remove that 1% hydrogen as a waste product containing no more than 0.74 TBq (20 Ci) per year of tritium which would be released with the hydrogen waste product. I have not discussed the tritium breeding blanket. Mr.Muhlestein in his paper (IAEA-TC-408/2)discussed work on lithium and lithium oxide and other materials that are being considered for the blanket. I would emphasize that the tritium that is bred in a blanket will contain a very large quantity of hydrogen impurity. There will be a lot of hydrogen bred in the blanket as well as tritium. The tritium that is extracted from the blanket system may contain as much as 50 or 60% protium. This will have to be separated from the tritium before tritium is placed into the fuel cycle. That can very simply be done in a destillation column system similar to what we are currently using for the isotope separation.

J.GRUBER: Did I understand it right that you will have atmospheric releases of 7.4 TBq (200 Ci) and a through-put of about one kg?

J.L.ANDERSON: In the TSTA Fuel Cycle we will recirculate this tritium at the rate of one kg per day. To do that, we will have a tritium inventory of approximately 200 g and our design objective is to release to the environment not more than 7.4 TBq (200 Ci) per year.

J.GRUBER: And what is the waste stream not released to the air but contained in oils etc.?

J.L.ANDERSON: We anticipate that solid waste that will be generated could contain several 37 TBq (thousand Ci) per year. But this is an estimate. In a few years we will have a better number on that. Certainly we would anticipate several 37 TBq (thousand Ci) per year of tritium contaminated solid wastes from the effluent processing streams and from our maintenance operations.

J.GRUBER: And no liquid waste?

J.L.ANDERSON: No liquid waste as such. Any liquid that would be generated would be collected on molecular sieve or other absorbents so that liquid waste disposal will not be considered. Liquid wastes will be further treated to solidify them or decompose them and recover the tritium.

L.D.MUHLESTEIN: You mentioned double wall piping and copper was the inner material. What is the outer piping material?

J.L.ANDERSON: We are using a variety of materials as the outer piping. In some cases it is copper, in some cases it is stainless steel and in some cases it is aluminium. We are trying a variety of materials to develop that system which would lend itself most readily to maintenance yet provide adequate containment. We are evaluating a number of materials for this. Our primary materials in the system are copper and stainless steel as the first containment. Any place that we operate at pressure below 2 atmospheres and at temperatures below 200° C we certainly prefer copper because one gets a lower amount of tritium permeation through the copper. When we have to deal with higher pressures and higher temperatures then our primary containment material is stainless steel.

L.D.MUHLESTEIN: The deuterium which is recycled goes to the injectors?

J.L.ANDERSON: Yes.

L.D.MUHLESTEIN: In a Tokamak a substantial amount of fueling comes from the injectors itself. So the gas that is injected into the Tokamak probably has to have a much higher tritium content than 50%. It might even be as much as 100%. Do you have any capability of separating the DT that comes off?

J.L.ANDERSON: In the isotope separation system with 4 columns we produce 4 hydrogen streams: the hydrogen waste, pure deuterium which can feed the system, a DT mixture and pure tritium. We can, therefore, provide deuterium for the injectors, we can provide a pure tritium stream and a one to one mixed DT. The deuterium-tritium containing hydrogen actually is the input from the vacuum system. The deuterium from the nuclear beam contains really very little tritium. We can inject it in the gas stream at the top of column 1, go right to the column where we separate H and run it to a stream to provide pure deuterium (see Figure A).

L.D.MUHLESTEIN: No. The D_2 goes back to the injectors and the T_2 goes to the Tokamak. What happens to the DT? How are you going to use it?

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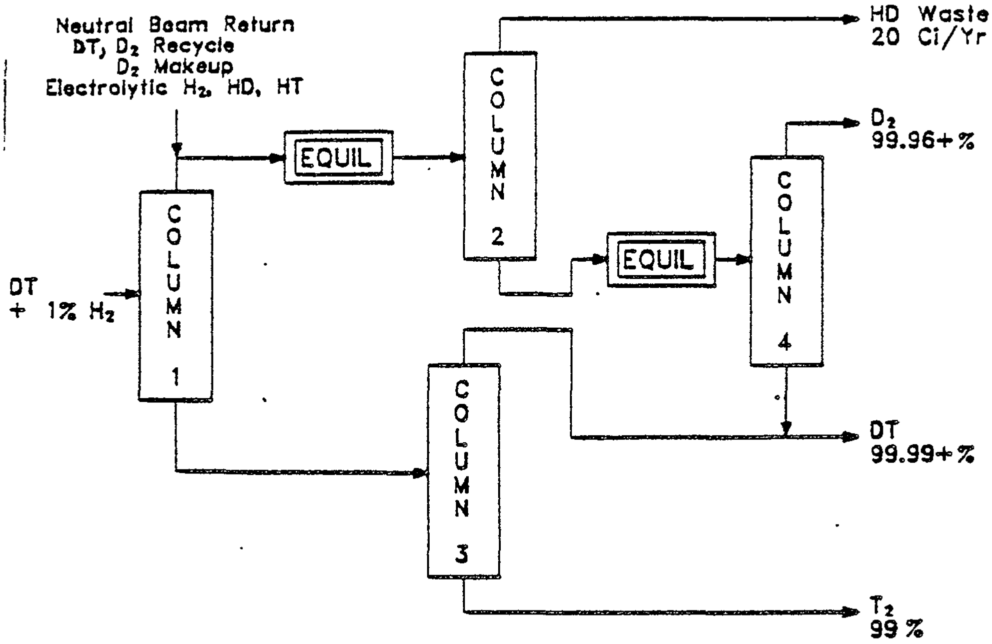

Figure A

J.L.ANDERSON: All of this goes to a storage area. And then it can be split - some to the injectors and some to the machine. Now this can also be recycled to the front end and we can continuously perform additional separation on this. This can always be recycled the way the system is planned, so that this component can indeed be permanently decomposed and end-up in these two streams.

L.D.MUHLESTEIN: That was my question. You got that built-in to recycle this.

J.L.ANDERSON: You have that capability so that you can, depending upon your input and your desired out-put, vary these streams by internal recycling.

J.G.CROCKER: You can also send that stream directly to your pellet fabrication and use it as DT pellets if you want.

T.K.ALLEN: The hydrogen isotopes in the exhaust system from a reactor are likely in atomic form. We are dealing here mostly with isotope on molecular form. I think adsorption and chemistry can be significantly different. If I remember the history of gaseous electrical discharges in the atomic form one certainly gets gas clean-up.Do you have any plans to ionize the gas or dissociate it in your solubilization chamber?

J.L.ANDERSON: We will investigate this experimentally later on. We looked at the amount of vacuum duct and the path that that gas will have to take from the reaction chamber to the vacuum pump. Indeed the gas is going to be pretty much in a molecular form and at ambient temperature.

T.K.ALLEN: I am trying to agree with that. My worry is the tritiation of the materials near or towards the chamber, the absorption of the material on the walls itself.

J.L.ANDERSON: This may be an area that is going to require considerably more effort after we get this primary process in hand. This is a point that we have considered and I guess we do not really know how serious that point is. I am not even sure that we know how to check it out and determine the extent of that problem. May be the answer will be found out from INTOR, FED, JET or TFTR.

J.L.ROUYER: Much of the work that you performed is similar to what is done in Valduc, France. I appreciate that you mentioned the getters that were explained at the San Diego Conference by experts from the Lawrence Livermore Laboratory. It is very important to try to trap tritium as tritium gas.

J.L.ANDERSON: I must emphasize that I do not feel very confident whether those getters will ever work. Livermore is still investigating these getters but they do not look very promising to me.

J.L.ROUYER: Concerning the general ventilation system: I appreciate the principles of dynamic confinement and the external zone having little higher than atmospheric pressure. If there would be a contamination of tritium between the second confinement and the tertiary confinement and the ventilation stopped, what would happen to the hirarchy of pressures?

J.L.ANDERSON: When the ventilation system is isolated and the operation of the detritiation system initated the air is processed to remove tritium: A portion of this air will be recycled to the room and a portion will be released to the environment. Only enough air will be released through the building stack to maintain the correct pressure balance.

J.L.ROUYER: I think that the principles are good but I would appreciate a discussion about the function of the detritiation system for the ventilation of the air of a room. It is rather expensive and we have to keep in mind future amounts of gas to be treated. In case of contamination would tritium fix to the walls. May be this reflects on concepts that minimize the amount of gas to be treated in such cases.

J.L.ANDERSON: We emphasize that in facilities we should go to compartmentalizations so that such a release would be into a smaller room instead of a large room as we had at TSTA. There have been many designs of fusion facilities, conceptual designs, that have included these room detritiation systems. However, nobody has built one and tested it. Within the military programmes in Europe and in the USA there has never been such a room detriation system built. This will be the first one that has been built and tested. Small scale experiments indicate that in order to minimize or reach acceptable levels of contamination on the wall one would need to process the air in a room at the rate of about 1% per minute to recover tritium before significant problems with residual contamination arise. Our system is designed to process the air at that rate of 1% per minute, but that is based on some small scale experiments. We will test our facility by releasing deuterium into the facility, spiked with a small amount of tritium, so that we can follow the clean-up using radiocounting techniques and we will see what the residual contamination problem will be.

J.L.ROUYER: Can your molecular sieves be regenerated?

J.L.ANDERSON: Yes, they are designed so that they can be regenerated and they can also be loaded out for disposal. We have both options.

W.DÄNNER: When you speak about injecting your hydrogen into the vacuum vessel, you do not think of pellet injection?

J.L.ANDERSON: Our responsibility at TSTA is to develop the tritium technology. We understand that indeed pellet injection is a means that is chosen for fuelling a reactor, that **that pellet** injection system must be qualified for use with tritium. In that case we would anticipate bringing a pellet injection system to Los Alamos and enclose it in a fuel cycle. We could then make pellets which we could inject then into the vacuum chamber. We would have to do that with any pellet injection machine to qualify it for use with tritium. But until such injection system is chosen we are not anticipating what it might look like and we are not building any machine until the physicists and the reactor designers chose an injection system. But once that is chosen we would include it in our loop at TSTA.

W. DÄNNER: You mentioned that the total inventory of tritium in TSTA is about 250 g and 100 g are located in the isotope separation part. You also mentioned that this component has a relatively high potential for release.

J.L.ANDERSON: Yes, this is all located in a relatively small volume.

W.DÄNNER: Did you also quantify the inventories in the other components and qualify the risk potential of these components?

J.L.ANDERSON: Yes, we looked at that. The figure showed a tritium inventory of 250 g. That inventory is now between 150 and 200 g. The inventory of TSTA is now less. Where will the inventory be: There will be between 100 and 115 g in the destillation column. There can be as much as 6 g in the vacuum system, because the vacuum pumps we are developing are cryogenic pumps and they will contain an inventory. We could have an inventory of 15 - 20 g in the fuel clean-up system where we remove impurities. These are the three areas where we would have significant inventories. The rest of the inventory will be in storage for the most part.

SUMMARY OF HEDL FUSION REACTOR SAFETY SUPPORT STUDIES

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ABSTRACT

The HEDL Fusion Reactor Safety Support studies are focused on characterizing blanket-coolant-material reactions for deuterium-tritium fusion reactor designs. The objective is to determine and examine potential safety and environmental issues associated with proposed blanket/coolant combinations under postulated accident conditions. The first studies considered liquid lithium as both blanket and coolant, and examined liquid lithium-material reactions. Liquid lithium reactions with oxygen, nitrogen, and various concretes have been characterized. Evaluations of lithium reaction extinguishment methods, lithium aerosol generation and collection, and the volatilization and transport of radioactive materials in connection with lithium-air reactions have been completed. Lithium compound blanket material reactions with water, a prime coolant candidate, have been characterized in terms of energy and gas release rates. Blanket materials considered were lithium aluminate, lithium oxide, lithium zirconate, lithium silicate, and lithium lead alloys (Li_7Pb_2 and $Li_{17}Pb_{83}$).

SUMMARY AND CONCLUSIONS

The conceptual designs of recently proposed U. S. deuterium-tritium fusion reactors have been strongly influenced by the desire to minimize safety and environmental issues. Major attention has been focused on minimizing the potential of releasing any radioactivity to the public and minimizing safety concerns for operating personnel. In this regard attention has been focused on identifying possible energy releasing chemical reactions with proposed blanket and coolant materials which may result following hypothetical accident sequences. Consideration has been given to coolant-blanket reactions, coolant-material reactions, blanket-material reactions, and coolant-coolant interactions. The HEDL Fusion Reactor Safety Support Studies have focused attention on lithium-material reactions, and lithium compound-coolant reactions. The lithium compounds being tested are lithium oxide, Li_7Pb_2, $Li_{17}Pb_{83}$, lithium aluminate, lithium zirconate, and lithium silicate with water as a coolant.

The lithium-material reaction tests provided containment atmosphere temperature and pressure responses resulting from a lithium pool or spray reaction in air and nitrogen atmospheres. Lithium concrete reactions were characterized. Volatilization and transport of radioactive metals during a lithium air reaction were identified. Successful lithium reaction extinguishment and lithium reaction aerosol control methods were demonstrated.

The lithium compound-coolant reactions are scoping in nature and to date have been limited to high temperature blanket materials added to water and water added to high temperature blanket materials.

The major conclusions from the program are:

- Liquid lithium can be safely handled under postulated accident conditions but special handling practices need to be considered.

- Lithium does react with both air and nitrogen to produce high flame and pool temperatures in air, and moderate pool temperatures in nitrogen. However, surrounding vessel temperatures are not extreme and the pressure is considerably less than some theoretical studies have predicted. For air atmospheres in a large cell, the pressure is dominated more by the gas consumption than the temperature increase.

- Lithium-concrete reactions are potentially more severe than similar sodium-concrete reactions and test results indicate it would be prudent to avoid lithium concrete contact.

- Major radioactive species such as cobalt, iron and manganese will be volatilized in a lithium-air reaction in contact with neutron activated stainless steel, and transported with lithium aerosols.

- Both large and small quantity lithium reactions can be extinguished. Powder applications are effective for small, shallow lithium pools, while space isolation and inert gas flooding using catch pans and/or cell liners are very effective for controlling large lithium spills.

- Large mass loading of lithium aerosols can be collected with conventional air cleaning systems with collection efficiencies larger than 99%.

- Lithium aluminate, lithium zirconate, and lithium silicate (at 600°C) do not appear to react with water. Li_7Pb_2, lithium oxide and lithium metal (at 600°C) react vigorously with water. $Li_{17}Pb_{83}$ at 600°C reacts mildly with water.

INTRODUCTION

Appropriate standards for developing commercially viable fusion reactors as an effective energy source are economics on one hand and safety and environmental acceptability on the other. Meeting the objective of developing an efficient, low cost energy source with socially acceptable safety and environmental impact usually involves trade-offs in terms of benefits derived versus acceptable risks. The conceptual designs of recently proposed U.S. deuterium-tritium fusion reactors(1) have been strongly influenced by the desire to minimize safety and environmental issues associated with operating reactors. Major attention has therefore focused on minimizing the potential of releasing any radioactivity to the public and minimizing safety concerns to operating personnel. Since many identifiable safety issues are closely coupled with the choice of coolant and blanket materials, considerable emphasis has been directed toward identifying coolant and blanket materials which provide acceptable risks.

A deuterium-tritium fusion reactor can be schematically visualized in simplified form as illustrated in Figure 1 where major elements such as the blanket material, primary coolant, heat exchanger, and secondary coolant are identified. In this schematic diagram one can identify areas where safety issues might be considered in terms of analyzing the results of hypothetical accidents beyond the design basis. In this context, general concerns would be in regard to coolant-blanket reactions, coolant-material reactions, blanket-material reactions, and coolant-coolant interactions which may result following some initiating accident sequence. The major concerns in analyzing these reactions are first, the energy and gas generation as they relate to the pressure which may be produced and which then may lead to violation of some containment boundary; and, second, the temperature produced from the reactions which may lead to volatilization of any radioactive isotopes contained in the materials. Figure 2 is a matrix which lists the leading coolant and blanket materials. From this matrix one can identify areas where data may be required to assist in safety analyses.

Liquid lithium is a candidate for both blanket material and coolant because of its physical properties and neutron absorption/tritium generation characteristics. For liquid lithium the areas of concern are coolant-construction material reactions and coolant-coolant interactions depending on the choice of the secondary coolant. For the lithium compound blanket materials listed in the matrix, with water as a coolant, the major area of concern is the blanket material-coolant reactions. Other coolants listed in this matrix are not included in this presentation because they are less important in the U.S. Fusion Program.

The objective of the HEDL Fusion Reactor Safety Support Studies is to provide information and data regarding liquid lithium reactions and lithium compound blanket material reactions which will be used as a basis for safety analysis in fusion reactors and support facilities. This program has two major tasks which are considered. First is the lithium spill studies, and second is the lithium compound-coolant material compatibilities studies.

LITHIUM SPILL STUDY RESULTS

The lithium spill studies were initiated late in 1977 with the objective of first completing a series of scoping studies to provide data regarding the properties of liquid lithium. These initial scoping studies have now been completed. The highlights will be presented in this paper and include the temperature and pressure from lithium pool and spray reactions in air and nitrogen atmospheres, a lithium concrete reaction sequence, volatilization and transport of radioactive metal isotopes associated with lithium-reactions, lithium reaction extinguishment, and lithium aerosol control.

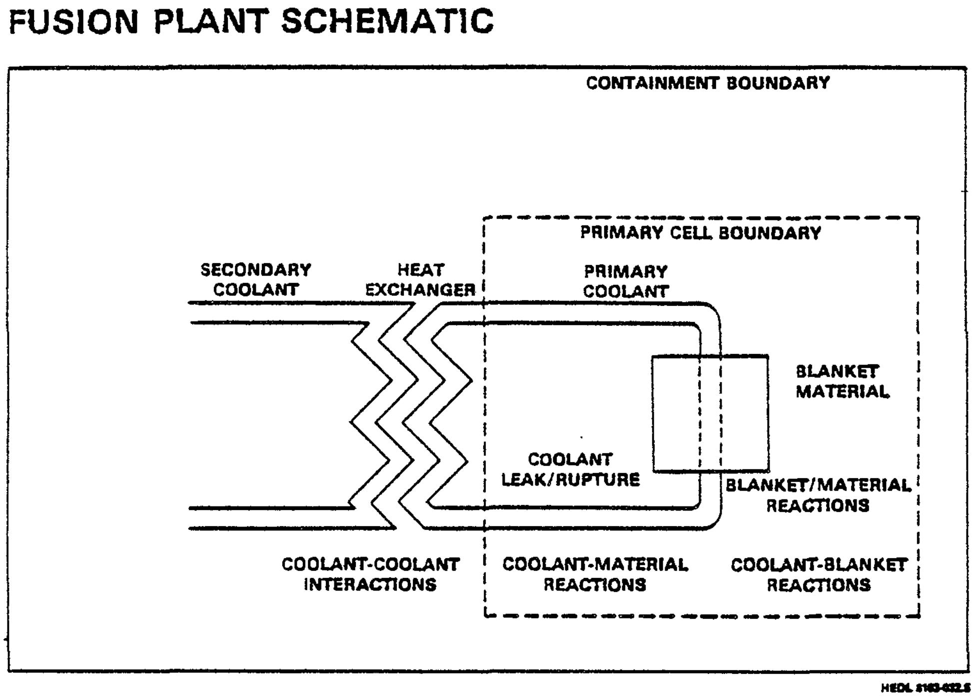

Figure 1. Simplified Schematic of Deuterium-Tritium Fusion Reactor.

Lithium Pool Reactions in Air/Nitrogen

The lithium pool reaction tests were performed in a 14 cubic meter vessel with test atmospheres of air or nitrogen. Ten kilograms of lithium at 230, 510, and 540°C was dumped into a 0.2 square meter pan and allowed to react. The test arrangements for the lithium pool reaction tests are shown in Figure 3.

The basic results of the lithium pool reactions in air are illustrated in Figure 4, where the lithium pool temperatures as a function of time are given for four test conditions. First is a normal air atmosphere with an initial lithium temperature of 230°C, the second is a normal air atmosphere with an initial lithium temperature of 510°C, and the third and fourth are

BLANKET MATERIALS	COOLANTS: LITHIUM	COOLANTS: WATER	COOLANTS: HELIUM	COOLANTS: SODIUM	COOLANTS: FLIBE	COOLANTS: KNO_3, $NaNO_2$, $NaNO_3$	CONSTRUCTION MATERIALS
LITHIUM		VERY REACTIVE			▨	▨	▨ CONCRETE AIR WATER
Li_2O		▨					▨
Li_7Pb_2		▨					▨
$Li_{17}Pb_{83}$		▨					▨
$LiAlO_2$		▨					▨
Li_2ZrO_3		▨					▨
Li_2SiO_3		▨					▨
OTHER		▨					▨
CONSTRUCTION MATERIALS	CONCRETE WATER AIR			▨ CONCRETE WATER AIR	▨	▨	☒
SECONDARY COOLANTS (WATER SODIUM)	▨ WATER			▨	▨	▨	

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Figure 2. Coolant-Blanket-Material Interaction Matrix.

LITHIUM POOL-ATMOSPHERE REACTION TESTS

TEST CONDITIONS
LITHIUM POOL
MASS - 10 kg
EXPOSED SURFACE - 0.2 m^2
CONTAINMENT VESSEL
VOLUME - 14.1 m^3
ATMOSPHERES TESTED
AIR
NITROGEN
HELIUM/NITROGEN
CARBON DIOXIDE
TOTAL TESTS COMPLETED 11

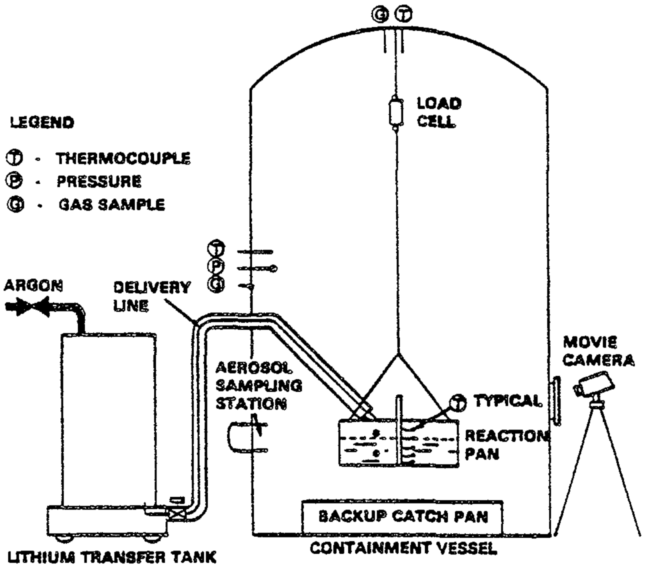


Figure 3. Lithium Pool-Atmosphere Reaction Test Conditions and Equipment Schematic.

a moist air atmosphere with an initial lithium temperature of 230 and 540°C, respectively. The maximum pool temperatures were approximately 1100°C, and were independent of either initial spill temperature or atmospheric conditions. Also illustrated in Figure 4, are the pressure and gas consumption which were measured for the normal humidity air test with the initial lithium pool temperature at 510°C. As lithium reacts with the air atmosphere, oxygen and nitrogen are consumed and heat is released. If one considers the pressure which would result due to the atmosphere temperature increase measured in the reaction chamber, one would see a pressure increase as illustrated in Figure 4. The calculated pressure change due to gas consumption is also illustrated in Figure 4. As can be seen, the actual pressure which was observed in the test chamber was dominated by the consumption of oxygen and nitrogen gas in the test cell.

Similar results for a lithium pool reaction in a nitrogen atmosphere are illustrated in Figure 5. Here the maximum lithium pool temperature measured was strongly dependent upon the lithium spill temperature. Also shown in Figure 5 are the pressure and gas consumption for the test with an initial lithium temperature at 840°C. It is noted that the pressure decreased in direct proportion to the amount of nitrogen gas which was consumed. Note the increase in gas consumption at a delayed time is caused by increased pressure due to the introduction of more nitrogen into the test cell to avoid a large negative pressure.

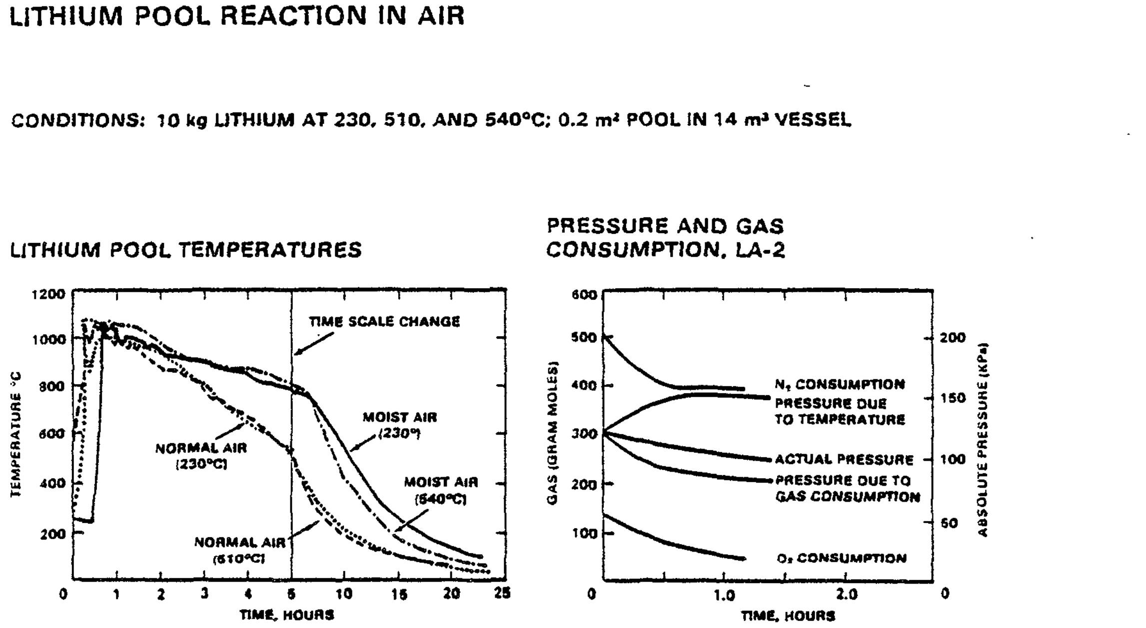


Figure 4. Lithium Pool Reactions in Air

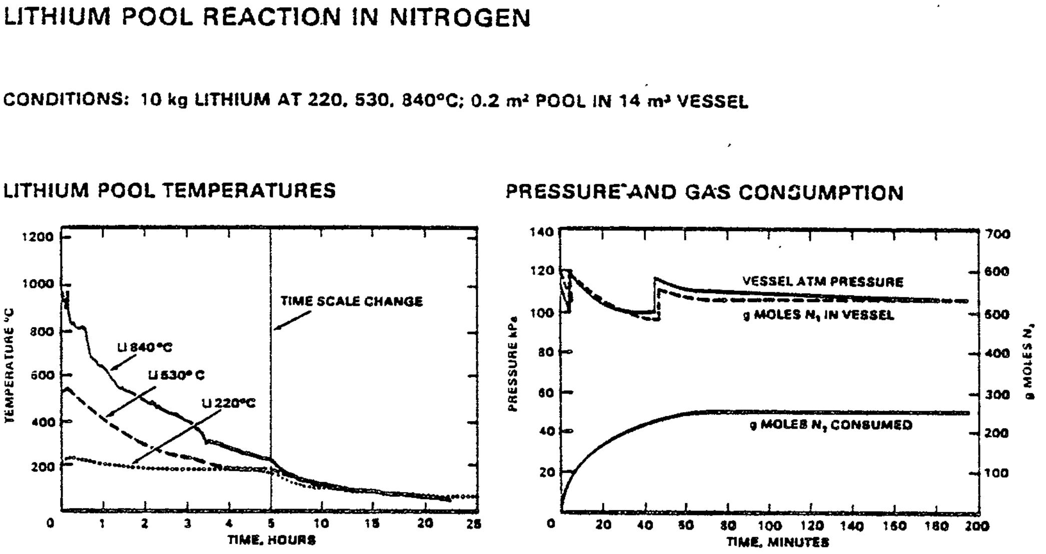


Figure 5. Lithium Pool Reactions in Nitrogen

Lithium Spray Reactions in Air and Nitrogen Atmospheres

Four lithium spray tests were performed in a 14 m^3 vessel where 1.5 to 2.4 kg of lithium was sprayed into the vessel at 1.4 to 4.9 grams per second at lithium spray temperatures between 425 and 650°C. The basic test conditions are illustrated in Table I. The test results for the lithium spray reactions in air and nitrogen are shown in Figures 6 and 7 respectively, where atmosphere temperature, pressure and gas consumption are shown as functions of time. For the low lithium temperature spray tests, LSA-1 and LSN-1, the major reaction was a pool fire at the conclusion of the spray which is indicated by the increased gas consumption at these times. For the high lithium temperature spray tests, LSA-2 and LSN-2, the spray reacted continuously as indicated by the gas consumption throughout the spray period. These tests indicate that the critical temperature at which a lithium spray in air and nitrogen atmospheres (under the specific test conditions) reacts to completion is between 425 and 650°C. The peak pressures produced for the high temperature lithium spray test in air and nitrogen occurred toward the end of their spray periods and amounted to a 31 percent increase for the spray in air and a 13 percent increase for the spray in nitrogen. The vessel atmosphere temperature and pressure increases generated for the spray in air are greater than for the spray in nitrogen because in air some of the lithium reacts with oxygen to release 285 Kcal per mole of O_2 consumed, while that which reacts with nitrogen releases only 95 Kcal per mole of N_2 consumed.

TABLE I

Lithium Spray Test Conditions

Test	Atmosphere	Volume m^3	Initial Atm		Lithium Spray				
			Temp°C	Pres KPa	Rate g/sec	Temp°C	Height m	Direction	Duration Sec
LSA-1	Air	14.1	25	110	1.59	425	3.4	down	600
LSA-2	Air	14.1	38	102	1.64	650	3.4	down	615
LSN-1	Nitrogen	14.1	18	108	1.36	427	1.5	up	1560
LSN-2	Nitrogen	14.1	36	109	4.9	649	3.4	down	307

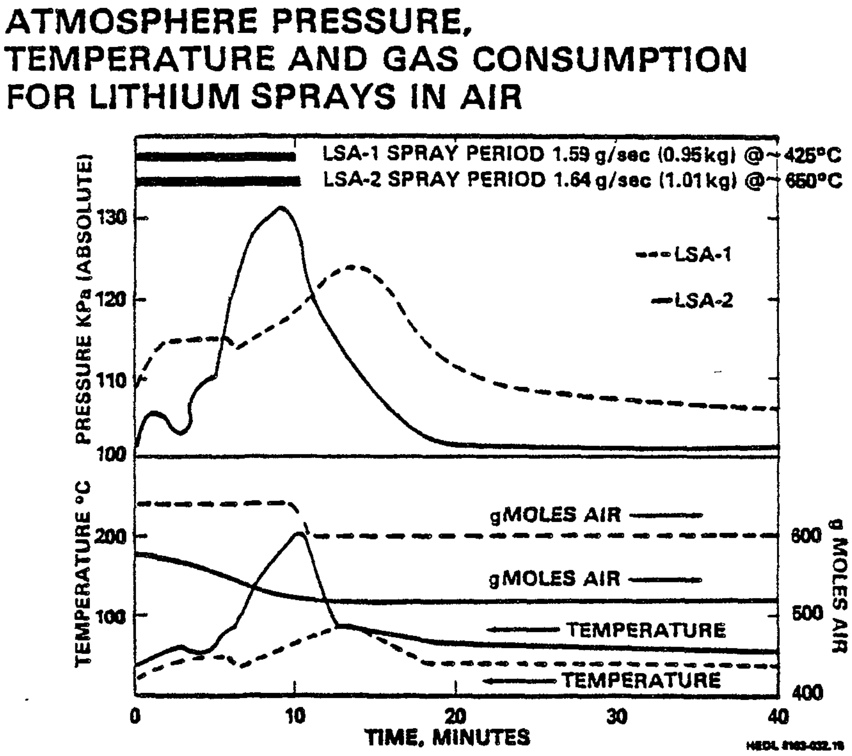


Figure 6. Atmosphere Pressure, Temperature, and Gas Consumption for Lithium Sprays in Air.

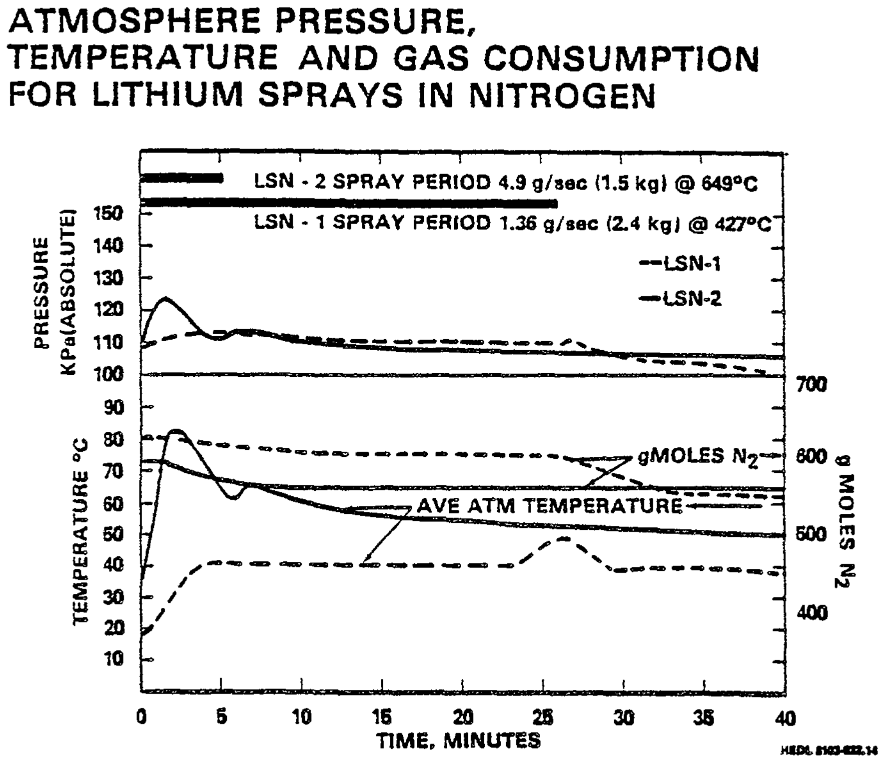


Figure 7. Atmosphere Pressure, Temperature, and Gas Consumption for Lithium Sprays in Nitrogen.

Lithium Concrete Reactions

The basic test arrangements for the lithium concrete reaction experiments are illustrated in Figure 8. Tests with basalt, magnetite and limestone aggregate concrete were performed. Only major conclusions from these tests will be reported here. The basic reaction sequence can be illustrated in Figure 9. As the hot lithium is dumped onto the ambient temperature concrete there is an initial thermal shock to the concrete. A surface reaction occurs between the lithium and surface water of the concrete and aggregate. The chemical heat of reaction and hot lithium then heat the concrete driving water as steam to the hot lithium-concrete interface. The released water reacts with lithium to produce hydrogen and generate heat. For initial lithium temperatures greater than 260°C, heat from the lithium-water reaction is sufficient to increase the temperature of the materials such that lithium will then react with the basic constituents (such as SiO_2 in basalt concrete, and Fe_3O_4 in magnetite concrete) to generate more heat and reduce the metal ions to elemental form. In this regard then lithium is more reactive with concrete than sodium. In the limestone concrete the predominant reactions are with water and CO_2.

Hydrogen is produced from the lithium-water reaction and it reacts to form lithium hydride which is collected in the lithium pool until either the pool is saturated or the lithium hydride decomposition temperature is reached. Hydrogen is released instantaneously as the lithium pool is heated to the LiH decomposition temperature. If the atmosphere above the lithium pool should be air then the hydrogen would be ignited by the lithium-air reaction and would burn.

From the tests performed it is concluded that in fusion facilities which use liquid lithium it would be prudent to avoid lithium-concrete contact such that these energy producing lithium-concrete reactions can be prevented.

LITHIUM-CONCRETE REACTION TESTS

TEST CONDITIONS
- CONCRETE TYPES
  - MAGNETITE
  - BASALT
  - LIMESTONE
- EXPOSED SURFACE AREA
  - 0.094 m^2
  - 0.28 m^3
- LITHIUM MASS 6.4 TO 15.9 kg
  - INITIAL TEMPERATURE 260°C TO 870°C
- ATMOSPHERE
  - ARGON
  - AIR

TOTAL TESTS COMPLETED 8

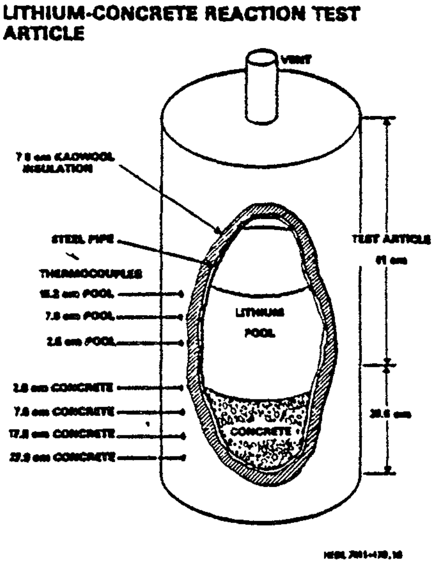


Figure 8. Lithium-Concrete Reaction Test Conditions and Schematic.

LITHIUM-CONCRETE REACTION SEQUENCE

1. SURFACE REACTION-LITHIUM WITH WATER AND AGGREGATE.
2. HEAT OF REACTION AND HOT LITHIUM HEATS CONCRETE DRIVING WATER TO HEATED SURFACE.
3. RELEASED WATER REACTS WITH LITHIUM LIBERATING HYDROGEN AND ENERGY.
4. LITHIUM-WATER REACTION SUFFICIENT TO PRODUCE ENERGETIC REACTIONS

$$\text{LITHIUM} + \left\{ \begin{array}{l} SiO_2 \\ Fe_3O_4 \end{array} \right\} \begin{array}{l} \text{- BASALT CONCRETE} \\ \text{- MAGNETITE CONCRETE} \end{array}$$

   FOR LIMESTONE CONCRETE PREDOMINANT REACTIONS ARE WITH WATER/CO_2

5. LIOH DOES NOT FORM; H_2 RELEASED FORMS LiH_2 UNTIL POOL SATURATED OR DECOMPOSITION TEMPERATURE REACHED
6. IN AIR ATMOSPHERE RELEASED HYDROGEN BURNS BECAUSE OF LITHIUM IGNITION SOURCE.

PRUDENT TO AVOID LITHIUM-CONCRETE CONTACT

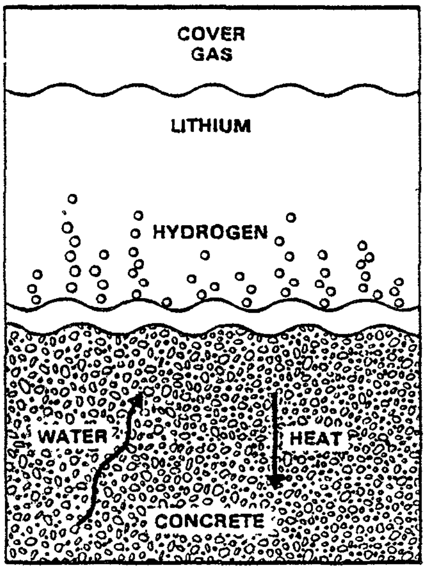


Figure 9. Lithium-Concrete Reaction Sequence

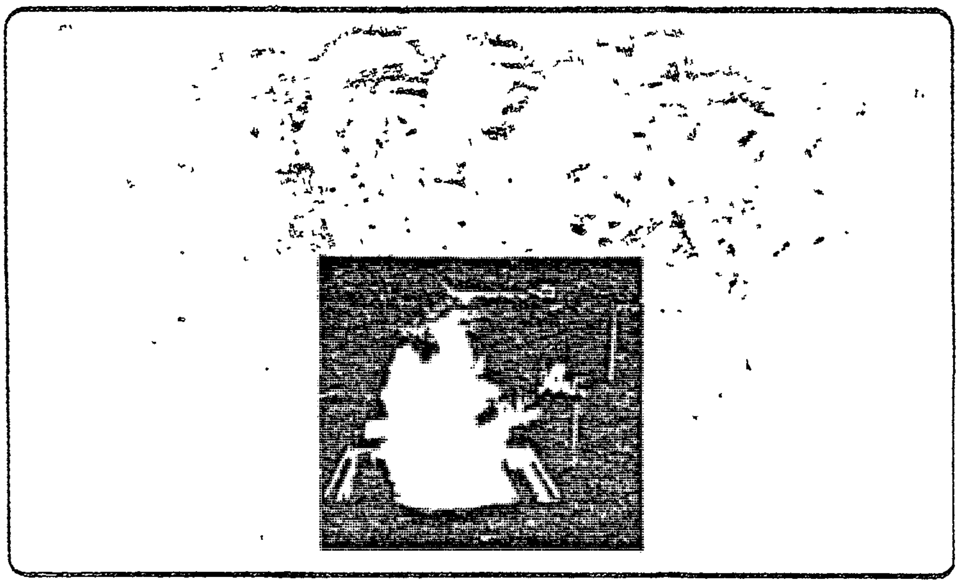
<u>Volatilization and Transport of Radioactive Materials</u>

One source of radioactive materials contained in a fusion plant would be the activation products created by the high neutron flux. These activation products could be found in the coolant material. Thus under a postulated accident condition, where the coolant boundary may have been violated, the radioactive materials may then be volatilized and transported with lithium aerosols which are generated by lithium-air reactions.

A scoping test was completed to determine the amount of radioactive metal volatilized and transported with lithium aerosols. Radioactive species which would be associated with first wall materials of 316 stainless steel, vanadium, and a titianium-zirconium-molybdenum alloy were tested. In this test 45 kilograms of lithium was reacted as a pool with an exposed surface area of 0.55 square meter. The test cell volume was 335 cubic meters. Elemental cobalt, iron, nickel, chromium, manganese, vanadium, lead, molybdenum, and zirconium were added to the lithium. Constituents from the 316 stainless steel were also corroded from the reaction pan by the reacting lithium. The plate-out and settling of the lithium aerosol was observed and the residue was examined by neutron activation analysis to determine the presence and quantities of the test metals. The test results indicate that cobalt, iron, and manganese were volatilized and transported with the aerosols. We therefore, conclude that if these materials were present in the coolant and the coolant boundary was breached, then they would be volatilized by a lithium-air reaction and transported with the resulting lithium aerosols. The major test results are illustrated in Figure 10.

VOLATILIZATION AND TRANSPORT OF RADIOACTIVE MATERIALS

- **TEST CONDITIONS:**
  - **45 kg LITHIUM: 0.55 m^2 POOL IN 335 m^3 CELL**
  - **TEST METALS ADDED: Co, Fe, Ni, Cr, Mn, V, Pb, Mo, AND Zr**
- **TEST RESULTS**
  - **COBALT,IRON AND MANGANESE ARE THE MAJOR METAL SPECIES OF CONCERN RELEASED FROM LITHIUM POOL FIRES**



HEDL 8101-181.8

Figure 10. Volatilization and Transport of Radioactive Metals.

Lithium Reaction Extinguishment

Another series of tests were completed to demonstrate effective techniques in extinguishing lithium reactions should they occur. There are several powders which exist for extinguishing lithium reactions. The test results demonstrated that carbon microspheres were more effective, but required 3.7 kg of carbon microspheres per kg of liquid lithium. The carbon microspheres were effective in extinguishing small or shallow pool reactions but were not effective for large or deep lithium pool reactions.

By far the most effective method for controlling lithium reactions is space isolation of the affected area followed by an inert gas purge. The basic results of lithium reaction extinguishment by space isolation and inert gas flooding can be seen from Figure 11 where the lithium pool temperatures are shown as functions of time. The first test was a lithium pool reaction in an air atmosphere where no attempt was made to extinguish the reaction. Note that pool temperatures above 1000°C were observed. Second, are the results where lithium was dumped into a catch pan which had a perforated cover. The maximum lithium pool temperature was reduced to 700°C. Third, are the results when the catch pan below the perforated cover was inerted with an argon atmosphere 15 minutes after the lithium spill, the maximum lithium pool temperature was further reduced to about 600°C. Note that the cooldown time of the lithium pool is also improved.

LITHIUM REACTION EXTINGUISHMENT

POWDER APPLICATIONS

- EFFECTIVE IN EXTINGUISHING SMALL, SHALLOW POOL REACTIONS AND NOT EFFECTIVE FOR LARGE OR DEEP LITHIUM POOL REACTIONS

SPACE ISOLATION/INERT GAS FLOODING

- CATCH PANS AND/OR CELL LINERS PROVIDE MOST EFFECTIVE METHOD TO CONTAIN LITHIUM SPILLS AND EXTINGUISH REACTIONS

LITHIUM POOL TEMPERATURES OF REACTION SUPPRESSION SUMPS

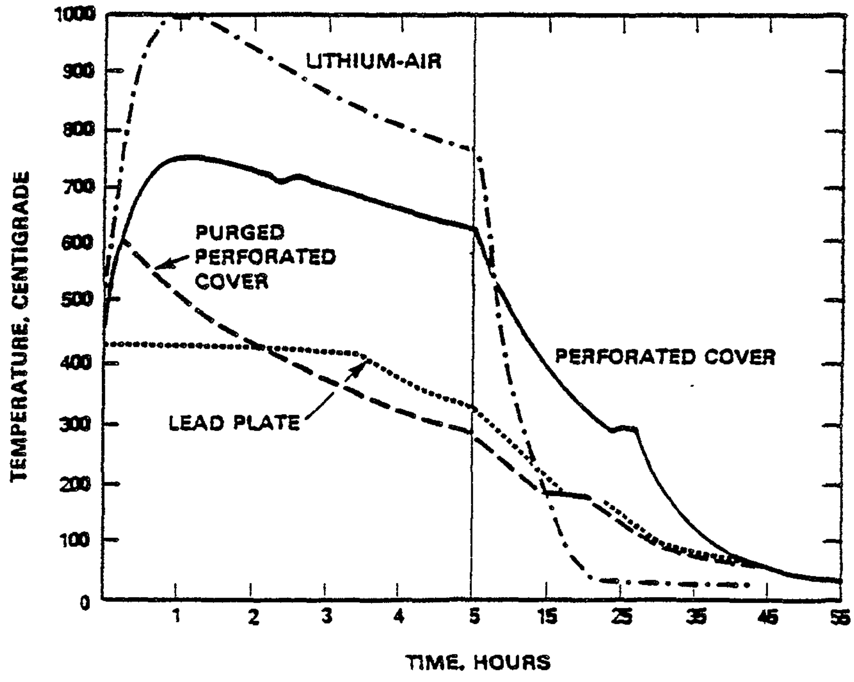


Figure 11. Lithium Reaction Extinguishment Test Results.

In a fourth test lithium was added to an inerted sump with a lead plate cover. The lithium melted the lead plate at point of contact and then flowed into the sump below. Fifteen minutes after the spill, the space in the catch pan was purged with argon. In this case, the lithium pool temperatures decreased from the initial spill-temperature.

From these experiments one can conclude that large quantities of lithium can be adequately controlled under accident conditions by isolating the spill area and inerting with argon gas.

Lithium Reaction Effluent Control

It has been demonstrated that radioactive materials will be volatilized because of high lithium reaction temperatures and then transported with lithium aerosols. Also lithium aerosols are quite toxic. For these reasons it may be necessary to control the release of aerosols resulting from a lithium spill. Several tests were performed using air cleaning devices such as sand and gravel bed filters, high efficiency particulate air filters (HEPA), and aqueous scrubber systems. These test demonstrated that large mass loadings of lithium aerosols can be collected with efficiencies larger than 99%. For an aqueous scrubber system the amount of aerosol collected is dependent upon the quantity of water available and the solubility of lithium aerosol in water. It is concluded that for both practical applications and emergency situations lithium aerosols can be collected and controlled.

LITHIUM COMPOUND-COOLANT MATERIAL COMPATIBILITY STUDIES

As will be recalled from Figure 2, the main emphasis of this phase of the program was to examine blanket material-water reactions for the lithium compounds lithium oxide, Li_7Pb_2, $Li_{17}Pb_{83}$, litihum aluminate, lithium zirconate, and lithium silicate. The results to be presented are initial scoping results.

Preliminary Information Base

The objectives of this phase of the program are to assemble available information concerning fusion reactor blanket-coolant-materials compatibility, and potential safety and environmental issues related to use of proposed lithium compounds as blankets. The initial focus is on completing a literature survey in relation to the reaction properties of lithium aluminate, lithium lead alloys, lithium oxide, and lithium zirconate. Also, design concepts, possible accident scenarios, and major postulated safety and environmental issues are being reviewed for promising fusion reactor designs.

Material Compatibility Scoping Studies

The objectives of this phase of the program are to provide material compatibility data for proposed lithium compound blanket materials and coolants which will be required to support material selection and safety studies. A series of scoping studies have been initiated to determine the reactions of lithium aluminate, lithium lead alloys, lithium zirconate, and lithium silicate with water. Two basic configurations were tested in an inert atmosphere. First, is the configuration where the blanket material at 600°C was added to water at 98°C. Second, is the situation where water at 98°C was added to the blanket material at 600°C. The basic test results for the hot blanket materials added to water under an open argon atmosphere are illustrated in Table II. The amount of blanket material was nominally 5 grams with 300 mililiters of water. The reactions of lithium metal added to water were also observed for comparison purposes. Lithium aluminate, lithium zirconate, and lithium silicate did not react exothermically with water. The major reaction observed for these compounds was heating and vaporization of water to produce steam. Lithium oxide produced a moderate exothermic reaction with the production of white aerosol. $Li_{17}Pb_{83}$ also produced a mild reaction but released some black aerosol; while Li_7Pb_2 produced a very exothermic reaction with a very dense black aerosol. The Li_7Pb_2 exothermic reaction was comparable to the lithium-water reaction.

TABLE II

BLANKET MATERIAL (600°C) ADDED TO WATER (98°C)

MATERIAL	QUANTITY BLANKET/WATER	BEHAVIOR
$LiAlO_2$	5 g/300 ml	NO REACTION
Li_2ZrO_3	5 g/300 ml	NO REACTION
Li_2SiO_3	5 g/300 ml	NO REACTION
Li_2O	4.5 g/300 ml	MODERATE REACTION
Li_7Pb_2	2 g/300 ml	EXOTHERMIC REACTION
Li_7Pb_2	5 g/300 ml	EXOTHERMIC REACTION
Li METAL	0.15 g/300 ml	EXOTHERMIC REACTION
Li METAL	0.5 g/300 ml	EXOTHERMIC REACTION

TABLE III

TEST RESULTS - WATER AT 98°C ADDED TO BLANKET MATERIAL AT 600°C

MATERIAL	QUANTITY BLANKET/WATER	BEHAVIOR
$LiAlO_2$	25cc/1g	NO REACTION
Li_2ZrO_3	25cc/1g	NO REACTION
Li_2SiO_3	25cc/1g	NO REACTION
Li_2O	25cc/1g	RAPID EXOTHERMIC REACTION
Li_7Pb_2	25cc/1g	EXOTHERMIC REACTION
Li METAL	25cc/1g	EXOTHERMIC REACTION

TABLE IV

SCOPING COMPATIBILITY TEST RESULTS FOR ENCLOSED SYSTEM

BLANKET MATERIAL	PEAK TEMP. (°C)	GAS RELEASED (MOLE/MOLE WATER)	HYDROGEN RELEASED (MOLE/MOLE WATER)	TIME TO REACH PEAK TEMPERATURE (sec)
LITHIUM ALUMINATE	600	1.0	0	–
LITHIUM ZIRCONATE	600	1.0	0	–
LITHIUM SILICATE	600	1.0	0	–
Li_7Pb_2	770	0.8	0.13	20
$Li_{17}Pb_{83}$	630	0.8	0.09	60
LITHIUM	900	0.37	0.16	10

The basic test results for water added to hot blanket material are illustrated in Table III. Nominally one gram of water was added to 25 grams of blanket material. Lithium aluminate, lithium zirconate, and lithium silicate again did not react. Lithium oxide, Li_7Pb_2, and lithium metal each demonstrated exothermic reactions. The qualitative scoping studies are now being repeated to obtain more quantitative information in terms of reaction energies and rates defined by temperatures and pressures which are observed. A series of these scoping tests have been completed by adding approximately 3 grams of water at 98°C to excess blanket materials at 600°C in an enclosed atmosphere. The reaction temperatures and quantities of released gas were measured. The blanket materials tested and test results are listed in Table IV.

REFERENCES

1. C. C. Baker, et. al. "An Overview of the Starfire Reference Commercial Tokamak Fusion Power Reactor Design," in Proceedings of the Fourth ANS Topical Meeting on the Technology of Controlled Nuclear Fusion, King of Prussia, Pennsylvania (October 14-17, 1980).

DISCUSSION

J.C.CHICKEN: The extinction powders that you mentioned are conventional fire fighting powders - nothing special?

L.D.MUHLESTEIN: Yes, that is right. One important item is that NaX should not be used on lithium reactions. NaX is basically sodium carbonate, and you end-up with a sodium fire. METAL-X really does not work well. Our recommendation is carbon microspheres are by far the most effective. In an unintentional spill at HEDL we used LITH-X. LITH-X becomes airborne and creates a worse problem than the lithium aerosols do. Our conclusion is to use carbon micropheres for small fires. I must emphasize that if some lithium is spread out on the floor over a large surface area you could put carbon microspheres on it and they would tend to extinguish the reaction. But if the lithium pool has any depth at all, the material will eventually sink to the bottom and the lithium continues to react. At some of the lithium systems at HEDL we have the capability for all cells to be isolated and inerted with argon. We think that is the best way. I know it is more expensive in a fusion facility, but this is the best way to handle a lithium reaction.

J.G.CROCKER: We did have some experience at Argonne where we had a small lithium handling loop and where a failure occured in one of the systems. The lithium spilled out and they used LITH-X. Typically it takes about 5 kg of LITH-X for every 1 kg of lithium that is spilled. For very large spills one does not have that amount of dry powder around to fight it.

L.D.MUHLESTEIN: In the HEDL incident the fire fighters obtained all the LITH-X we had, and then they obtained all the carbon microspheres. This was a deep pool in a tank continually being heated by the reaction. Again, the best way is to inert the reaction area with argon.

J.G.CROCKER: One needs to figure where the likely places for spills are and have routing of those spill areas to a sump. One needs to have the capability for inerting the atmosphere in that particular sump and then on top of that one should probably have some carbon microspheres around, in case one gets a spill where one does not necessarily want it.

L.D.MUHLESTEIN: I might point out that we have another large test coming-up, probably in a month, where we have a variety of sump arrangements. What we are trying to do there is look at designs that one might use for remotely extracting the reaction products, considering that they might be radioactive, put in another sump arrangement, and take your plant back to operation, or drain the lithium into a sump arrangement/catch pan that could be remotely removed. We think it will be very effective to inert the sump with argon atmosphere. The point is that one can really lower the reaction temperatures by inerting with argon. The high temperature leads to volatilization of radioactive products that may be contained in the coolant. If one can lower these reaction temperatures, and keep the material from reacting, one should not get volatilization products. Hence the radioactive transport should be greatly reduced.

J.G.CROCKER: The other factor that appears to be self-mitigating is that one will not generate the pressure within the containment to drive a leakage through the containment building. That is the problem with a fission reactor, where if one gets a pipe break, one gets the very high pressures being built-up in the containment. But here a lithium fire actually has a slight decrease in pressure. We have an analytical programme that goes along with this. We have a code development effort that is co-ordinated with the lithium spill work and we are calculating some results there. Most of the spill tests that we have been doing to-date have been with 10 kg of material. Later on in the year we are going to do a larger spill test with 100 kg. We are going to increase the quantity of material that we are spilling by an order of magnitude. One does not want to spend all the time working with 10 kg of material when one finds out that one gets a completely different answer if one is working with 100 kg of material. Reactors are not going to have only 10 kg. If the results of the 100 kg spill test are similar to the 10 kg tests then it is not necessary for us to go to larger spills. On the other hand if we get a significant change in result then we may have to factor in even larger spills later in the programme.

L.D.MUHLESTEIN: I should point out that the larger spills of 100 kg will be in a large test cell, with a large containment volume (about 8m in diameter by 20m high). The volume is about 900 m^3. The spill will be over a much larger surface area. With 100 kg we will be looking at pressures and temperatures generated by the reaction and compare them with the LITFIRE code predictions. We hope that this then will be a validation test for the LITFIRE computer code.

F.N.FLAKUS: Did you make studies where you added test metals such as cobalt or iron to the concrete?

L.D.MUHLESTEIN: No, we have not completed tests where these materials were added to the concrete. But I should point out, that at least for 316 SS, the majority of the volatile materials in our test results came from the reaction of the lithium with the stainless steel. Lithium actually corrodes stainless steel. We did not however add materials to concrete.

J.L.ROUYER: You tested volatilization of cobalt etc. Do you have a basis of comparison for sodium fires?

L.D.MUHLESTEIN: Not a quantitative comparison, but a qualitative one. It is a basic assumption in fission safety analysis that radioactive materials will coagglomerate with sodium aerosols and be transported with the sodium aerosols. That basic assumption is being tested in the USA programme basically at Oak Ridge. Some large scale aerosol behaviour tests have been completed in our facility.

J.L.ROUYER: But there is no indication that for sodium there is a preferential volatilization of cobalt?

L.D.MUHLESTEIN: Not a preferential volatilization if I understood you correctly. With these particular tests with lithium we were very much concerned with those species that had the highest BHP. That was iron, cobalt, manganese; and they were transported by the lithium aerosol. That was the issue we were considering.

J.L.ROUYER: Are there the same phenomena for lithium fire and sodium fire?

L.D.MUHLESTEIN: That will be correct.

J.L.ROUYER: Did you test for prefiltration in your filtration devices? Did you test acoustic agglomeration?

L.D.MUHLESTEIN: No, we did not with lithium. The reason is that prefilters are second order devices in liquid metal systems. There is some information that indicates that turbulent agglomeration may be effective for a cell when it would be desirable to have material quickly plated out on the surface before it starts to leak out. There are some data from the Havard Air Cleaning Laboratory on very small scale tests which indicate that turbulent agglomeration might be effective. We intend to look at turbulent agglomeration for the breeder programme because there it may be effective. For fusion facilities we do not think that other devices are necessary because standard systems, like an aqueous scrubber system, are very efffective and will contain very large mass loadings of aerosols. I need to point out that the amount of material which can be collected, is only dependant upon the solubility. In the case of lithium, the solubility of lithium aerosols in water is smaller in comparison with sodium. So in a comparable situation one could collect more sodium aerosols than lithium aerosol because of the difference in the solubilities.

J.L.ROUYER: In this case you need more liquid.

L.D.MUHLESTEIN: That is correct.

J.L.ROUYER: Is'nt it a large system?

L.D.MUHLESTEIN: No. If you consider the quantities of material that may be available in a fusion facility, for example, you would be talking about perhaps two liquid tanks with each about 6 meter in diameter. You are not talking about big systems like you would for the comparable sand and gravel bed filter. For example when you use a HEPA filter you need many banks. But the aqueous filters are rather small. In fact at HEDL we have developed and proof-tested a continuous aqueous cleaning sand and gravel bed which is very effective, very small, and which handles large quantities.

J.G.CROCKER: There was somebody at ISPRA who was interested in doing some liquid metal safety work with lithium there. He had not started any such work but it was in the programme. Does anybody know what the status of that is?

J.SCHNEIDER: Yes, Mr.Heinz Kottowski. He has been involved in sodium work for fission reactors. He has available at ISPRA 100 kg of lithium. There is a plan of work but this work has not been started yet.

SAFETY METHODOLOGY AND RISK TARGETS

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Abstract

In assessing the potential safety concerns of fusion, the experience from other energy sources lead to a variety of safety assessment approaches. The available approaches are:

(1) The maximum possible accident approach
(2) The maximum credible accident approach
(3) The probabilistic total risk assessment

In the first approach, the mechanistic development of the events leading to the safety concern is ignored. Instead, the total radioactivity of the plant is assumed accessible to the public. Such an approach is obviously conservative and unrealistic. In the second approach a selection is made among the most severe of the possible accidents, and the progression of the accident is modeled as mechanistically as possible. In this case, the passive and active accident mitigation capabilities of the plant are taken into consideration. The result is expected to be that none or only a fraction of the total radioactivity can be released to the public. The adverse effect of this approach is to concentrate attention on a particular accident class, and perhaps not allow for other classes, a judgement that may later become undesirable.

The probabilistic risk assessment requires the safety analysts to consider all classes of accidents and estimate both the probabilities of their occurrences and their consequences. Thus, the plant design in fact is subjected to a thorough investigation and the impact of alterations in design can be reflected in the total risk estimate. The disadvantage of this approach lies in the absence of well defined acceptable risk criteria as well as the large effect of public perception factors on the accepted risk.

This paper will review the impact of application of these approaches in determination of the level of protection needed against activation product release to the atmosphere.

Safety Methodology and Risk Targets

All forms of energy production are associated with some cost to society in the form of the potential for adverse health effects to the public as well as the workers involved in every stage of the power plant construction and operation. A proper design goal of any power plant is to minimize the environmental penalty and to insure that it lies within the boundaries of societal acceptance.

One of the main concerns of fusion power development is the potential radiological hazard. The radioactivity in a typical fusion plant will lie in four basic forms:

1) Tritium in both active and passive parts of the fuel systems.
2) The induced radioactivity in structural materials including first wall, blanket, shield, etc.
3) Coolant activation products and structural corrosion products circulating with the coolant, liquid breeder, or helium purge stream
4) Gaseous activation products, mostly resulting from activation in the gases in the plant building atmosphere.

In assessing the potential safety concerns due to releases of radioactivity from a fusion plant, the experience from other energy sources leads to a variety of approaches. These are:

1) The maximum possible accident approach
2) The maximum credible accident approach
3) The probabilistic total risk assessment approach.

In this paper, a brief review of the premise and shortcomings of each approach is presented, as well as the impact of application of these approaches in determination of the level of protection needed against radiological releases to the atmosphere. In actuality, the approaches sometimes overlap and the field of fusion safety can be seen as evolving from the first towards the third.

The Maximum Possible Accident

In the first approach, it is assumed that the total radioactivity of a plant is accessible to the public due to unspecified circumstances. This maximum possible accident is not analyzed mechanistically, so that the chain of events that may produce total accessibility is ignored. There are some attractive features to this approach. It is simple to consider and hence can be applied in the absence of concrete plant designs. Also, it forces the plant designer to consider protection systems against the most conservative estimates of accidental radioactivity releases. Hence, the safety measures that will be implemented will not be "overburdoned" by less severe accidents.

However, the total radioactivity in a plant will almost certainly not be released to the atmosphere as a result of a realistic accident. Hence, this approach does not provide the designer with the means to tailor the protection systems against the more probable releases. Furthermore, it could lead to erroneous decisions about the potential hazards of designs that may involve somewhat higher amount of radioactivity but with lower probabilities for suffering failure during unexpected events (accidents). Finally, the approach does not consider, and fails to guide, the actual design of reactor systems.

Because of the simplicity and limited need for data, this approach has been often applied to fusion plants. In most cases, the activity (in curies) and/or the biological hazard potential (BHP, measured in units inversely proportional to the MPC, the maximum permissible concentration for each isotope in air or water) is calculated for different structural materials and tritium breeders. A most recent example of the maximum possible accident approach is that of Holdren [1] where a thermodynamic treatment calculation of adiabatic flame temperature in a lithium fire ($\sim$2100 °C) leads to the assumption of nearly total release of structural activity.

The Maximum Credible Accident Approach

In the second approach, a selection is made among the most severe of the possible accidents, and a progression of the accident is modeled as mechanistically as possible. In this case, the passive and active accident mitigation capabilities of the plant are taken into consideration. Thus thermodynamics is combined with heat transfer mass flow, and chemical reaction kinetics to conservatively model the accident.

The approach has several advantages. A more complete picture of the history of and accident and its potential consequences are obtained. As such, unrealistic, over-conservative prediction of consequences is avoided. The impact of design choices on accident consequences can be examined. Finally, the modeling and identification of critical parameters helps determine future safety research needs

The approach has drawbacks. By nature it is more complex than the first approach and requires more information about the plant, making it more difficult to implement. The sophistication and accuracy of the modeling is often limited by the available experimental data. Since actual risk to the public is given by the consequence times the frequency of occurrence, and this approach neglects probabilities, an incomplete picture of the total risk is obtained. Also, the approach may concentrate attention on a particular accident class, and perhaps not allow for other classes, a judgement that may later become undesirable.

Several examples of the maximum credible accident approach exist. For instance, a continuing effort is underway at MIT to model the impact of the reactions concerning lithium compounds [2,3]. The LITFIRE (modeling lithium-air reactions) code basic model is shown in Fig. 1. A spill of lithium from one loop of UWMAK-III in the air filled containment is predicted to result in the time-temperature history shown in Fig. 2 [2]. As data and modeling have improved, the severity of the predictions is seen to have been reduced. Thus a detailed treatment of the lithium fire problem indicates that maximum temperatures are far below the simple adiabatic case. Figure 3 shows the model for an option allowing lithium compound-water reactions inside the blanket. Figure 4 and 5

[3] show the time-temperature histories for a sample case for the reaction zone and first breeder element response. Such mechanistic tools are also vital for the total risk assessment approach.

The Probabilistic Total Risk Assessment Approach

The third approach requires the safety analysts to consider all classes of accidents and estimate both the probabilities of their occurrence and their consequences. Thus, the plant design is subjected to a thorough investigation and the impact of alterations in design can be reflected in the total risk estimate. The total risk is the summation over all accidents of their probability times consequence.

The advantage of this approach is that the plant design is well digested from the safety point of view. The event and fault trees are the major tools of this analysis. The actual contribution of various accident sequences to the total risk is known so that a designer can concentrate on minimizing the overall risk by lowering either the event frequency or the potential consequences.

The approach has the drawback of high complexity and necessary data needs (e.g., specific detailed plant design). Furthermore, there is the absence of well-defined acceptable risk criteria by the public, mainly due to the large effect of public perception factors on the accepted risk.

The total risk approach has found some current application in the U.S. safety program by addressing the TSTA design [4], and deriving tolerable limits of structural radioactivity releases. Working backwards from a criterion that fusion risk should not exceed the WASH-1400 curve for fission and incorporating models for structural volatization and lithium fires, a target for maximum accidental release probability has been established and is shown in Figures 6 and 7. The fraction of the first wall that would be mobilized under realistic lithium fire conditions was found small compared to the tolerable limits [5].

Recently, an assessment was made of the risk to human health implied in the utilization of fusion reactors as a source of electricity [6]. This risk was compared to that of conventional and unconventional alternative energy sources.

While several approaches to such an assessment appear possible, the one chosen here is the summation of the total effects of all activities involved in the power cycle. In this method the risk is measured in terms of injuries, diseases and deaths of workers and the general public resulting from the following components of the fuel cycle:

1. Fuel raw material acquisition and processing ,
2. Fuel material transportation
3. Plant component fabrication, including emission
4. Plant construction
5. Plant operation and maintenance occupational effects
6. Plant operation public effects

Two reference designs, UWMAK-II and NUWMAK were used in the risk assessment. Both are conceptual designs for TOKAMAK fusion reactors, which is currently the leading type in reactor development. Table 1 contains the results of the assessment. Comparison of the two sets of figures indicates that the total man-days lost per megawatt electric produced by the plant per year (MDL/MWyr) are expected to be somewhat higher for UWMAK-II than NUWMAK. This is primarily due to the larger volume of the plant associated with UWMAK-II, which gets reflected in a higher risk in the categories of material acquisition, transportation and component fabrication.

It is assumed in this analysis that these plants will release 10 Ci/day to the atmosphere. The consequences of this release are the major source of the public health effect shown on Table 1 and is based on earlier studies [7]. Accidental atmospheric releases of radioactivity are assumed to be of a low probability due to both inherent and design factors.

Comparison to risk from coal, LWR, solar thermal and solar-photovoltaic plants has been undertaken. Three major sources of data on risk were used: the revised Inhaber study [8], Holdren et al. [9] and IAEA/IASA study [10]. The results are shown in Figure 8. It is seen that, compared to other fuel cycles, fusion is in a favorable position with respect to risk. The material requirements of the fusion plant are roughly comparable to the coal and LWR plants, but substantially below those of the renewable resources. Coal also has a higher risk component in fuel acquisition and transportation. On the other hand, provided that the assumptions of the present study are correct, the operational releases of a fusion plant are associated with smaller risk to the public than coal or LWRs. Hence, it is concluded that if fusion reactors can be developed in a manner that eliminates any significant risk contributions from accidents and waste management, the total health impact on society may be less than that associated with other fuel cycles.

Summary

Three approaches to safety assessments of fusion facilities have been discussed, including the advantages and disadvantages of each. Of particular note is the usefulness of the maximum credible accident approach and the total risk approach to the development and design processes.

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Table 1

Risk Components for UWMAK-II and NUWMAK (Ref. 6)

(Man-Day lost/MWe-yr)

	NUWMAK	UWMAK-II
Material acquisition	0.0897	0.3145
Component fabrication	0.254	0.402
Plant construction	0.94	0.977
Operation and Maintenance	.38 - .41	.38 - .41
Emissions	.0254 - .763	.028 - .8442
Transportation	.182 - .572	.399 - 1.256
Public health risk	.0113 - .610	.0113 - .610
	1.89 - 2.64	2.512 - 4.82

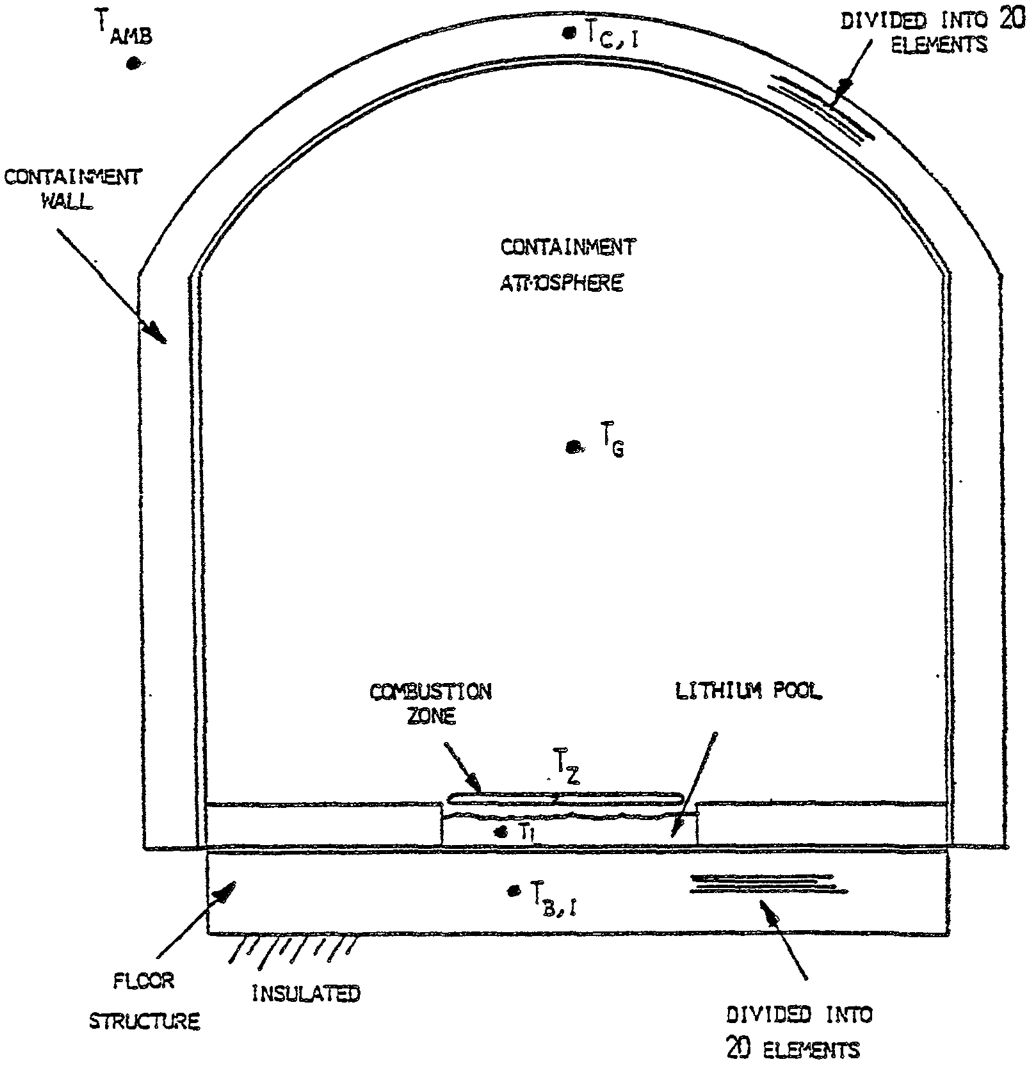

Figure 1: Basic Containment Structural Model

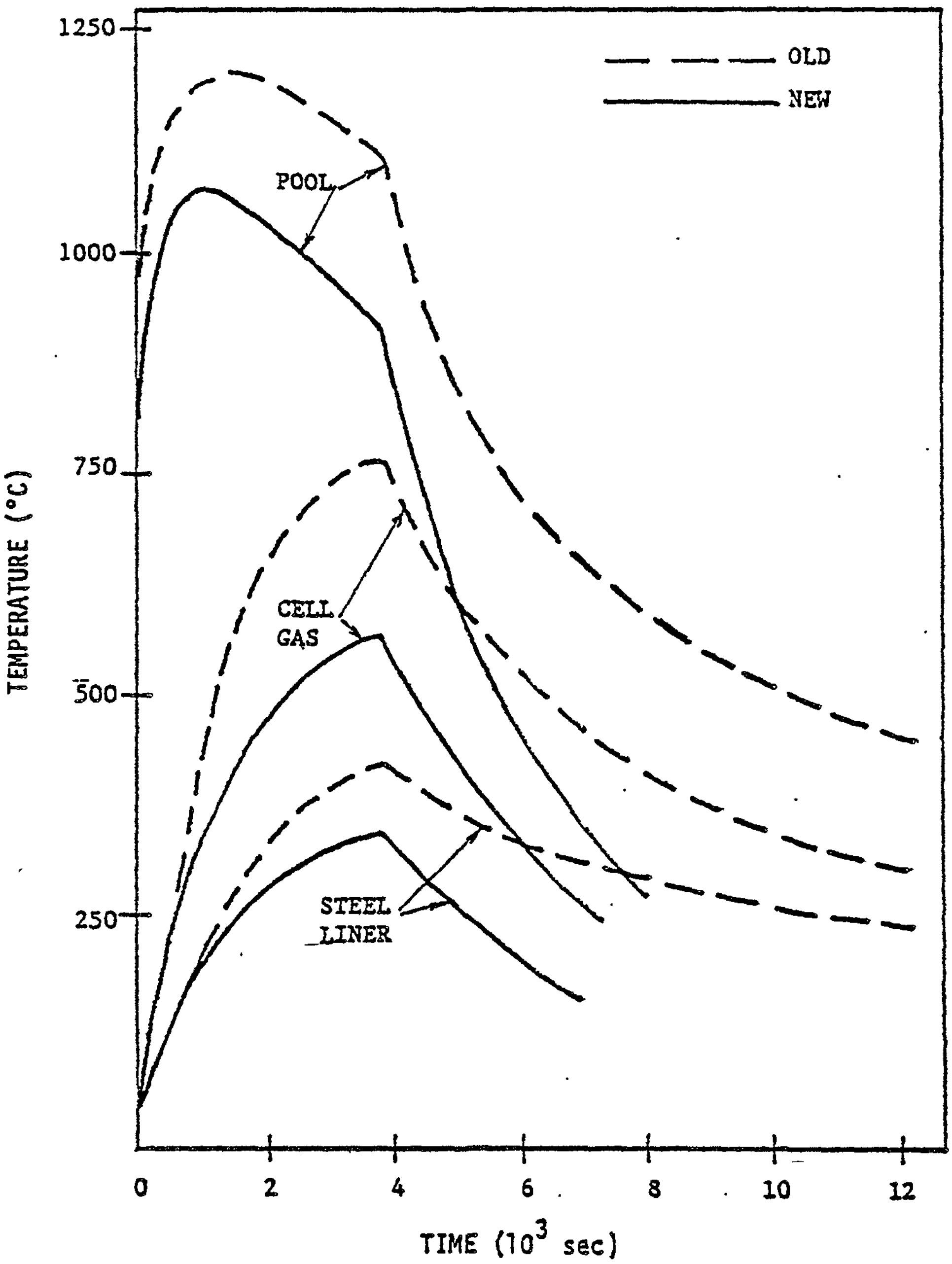

Figure 2: Comparison of Current LITFIRE Predictions with July 1978 Predictions.

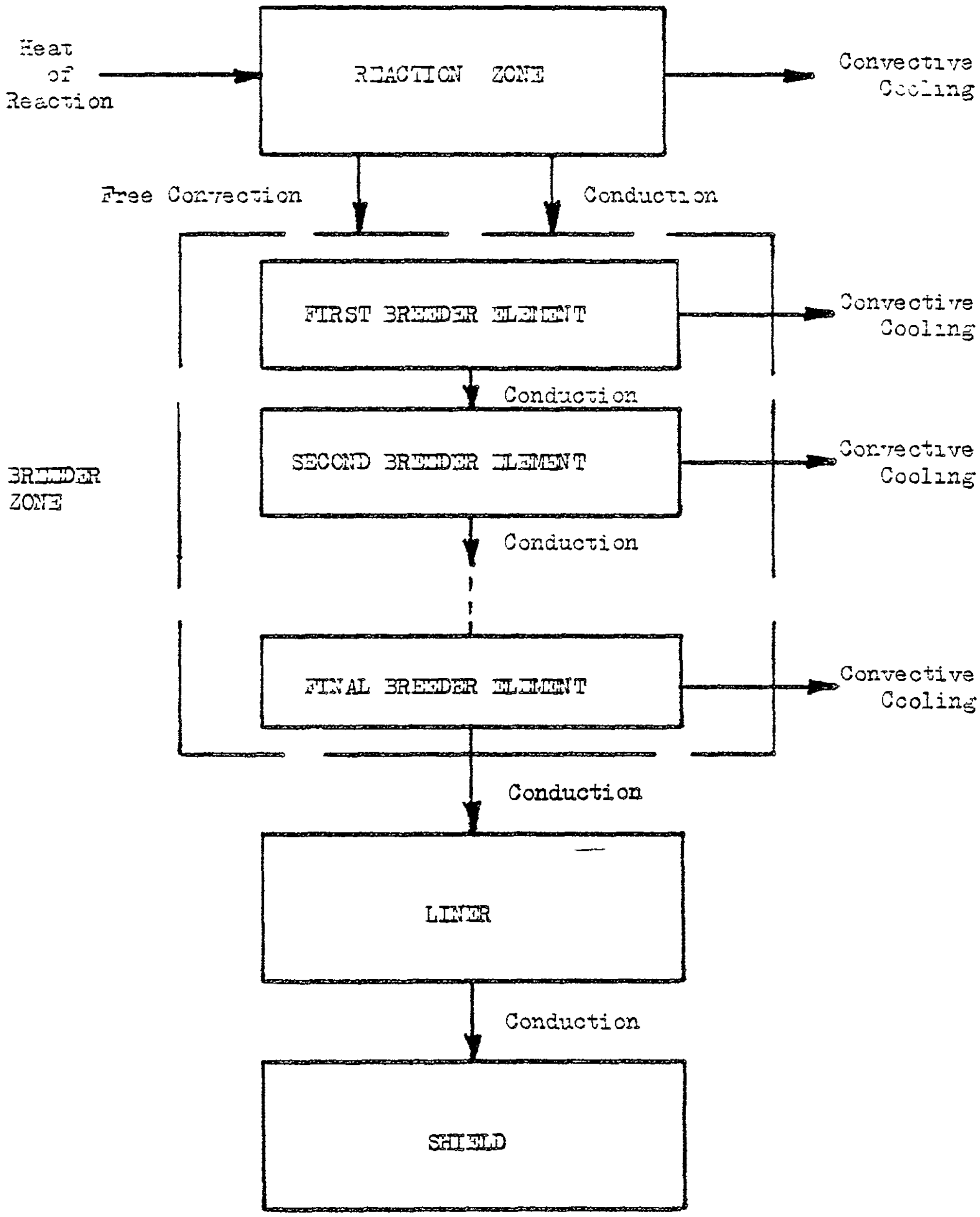

Figure 3: Internal Blanket Accident Option Heat Flow Diagram

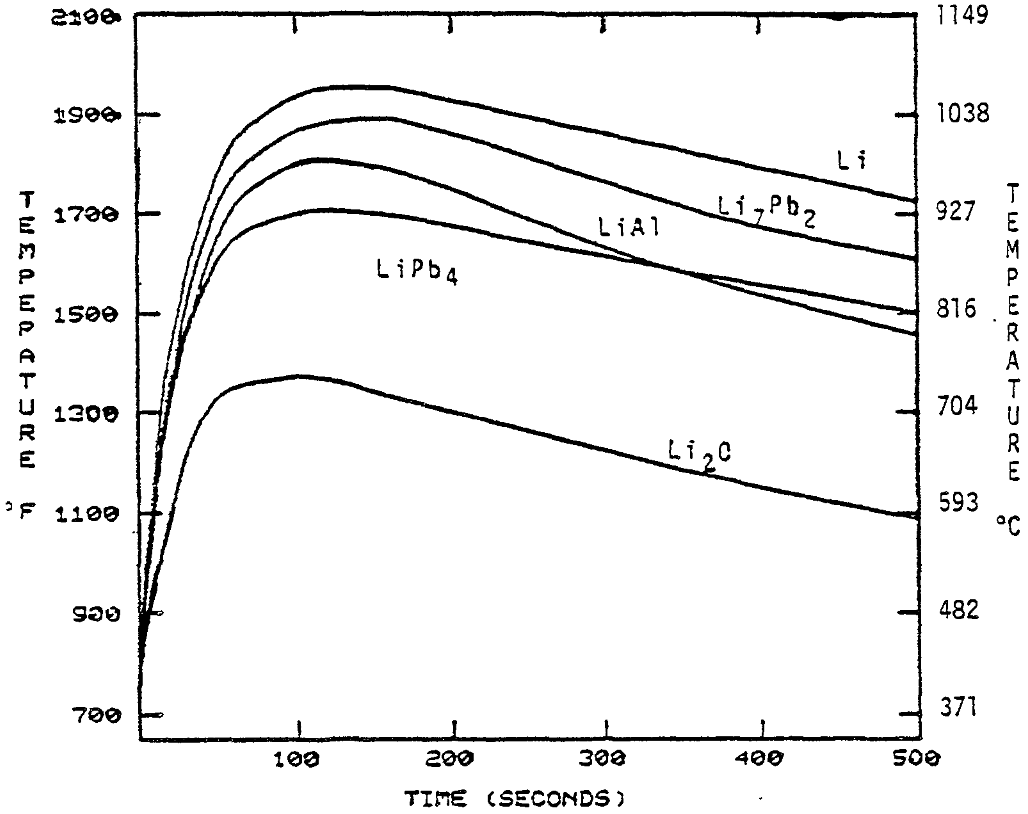

Figure 4: Comparison of Reactor Zone Temperature Profiles of the Various Breeders.

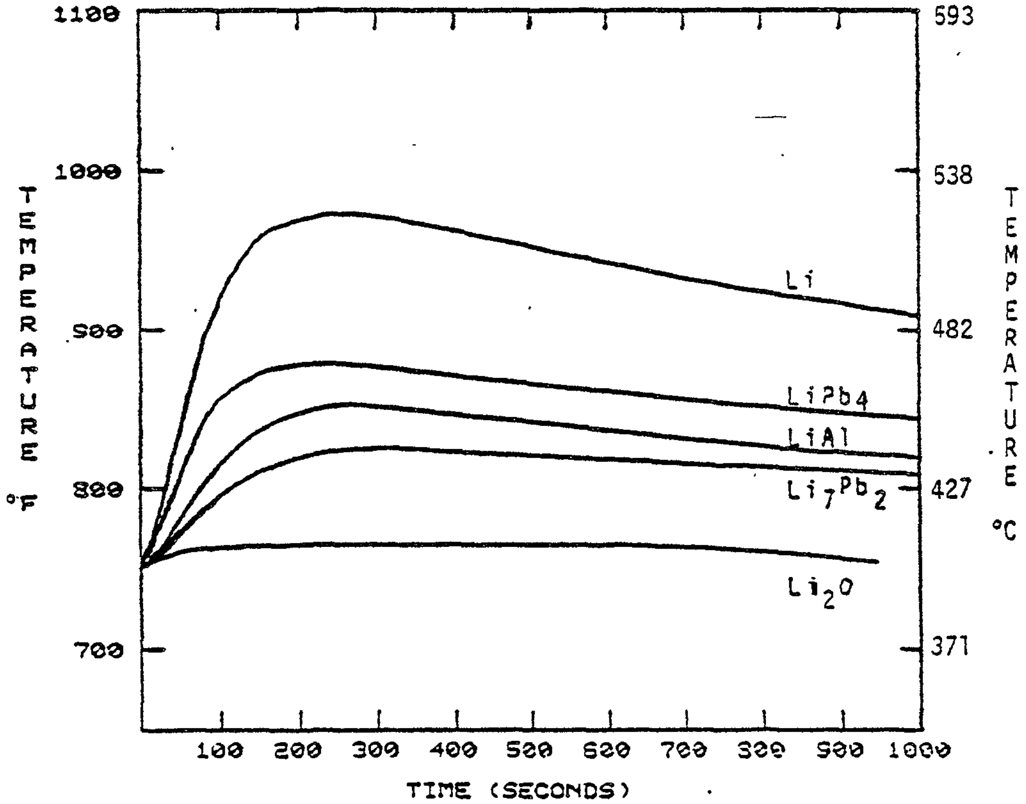

Figure 5: Comparison of First Breeder Element Temperature Profiles of the Various Breeders.

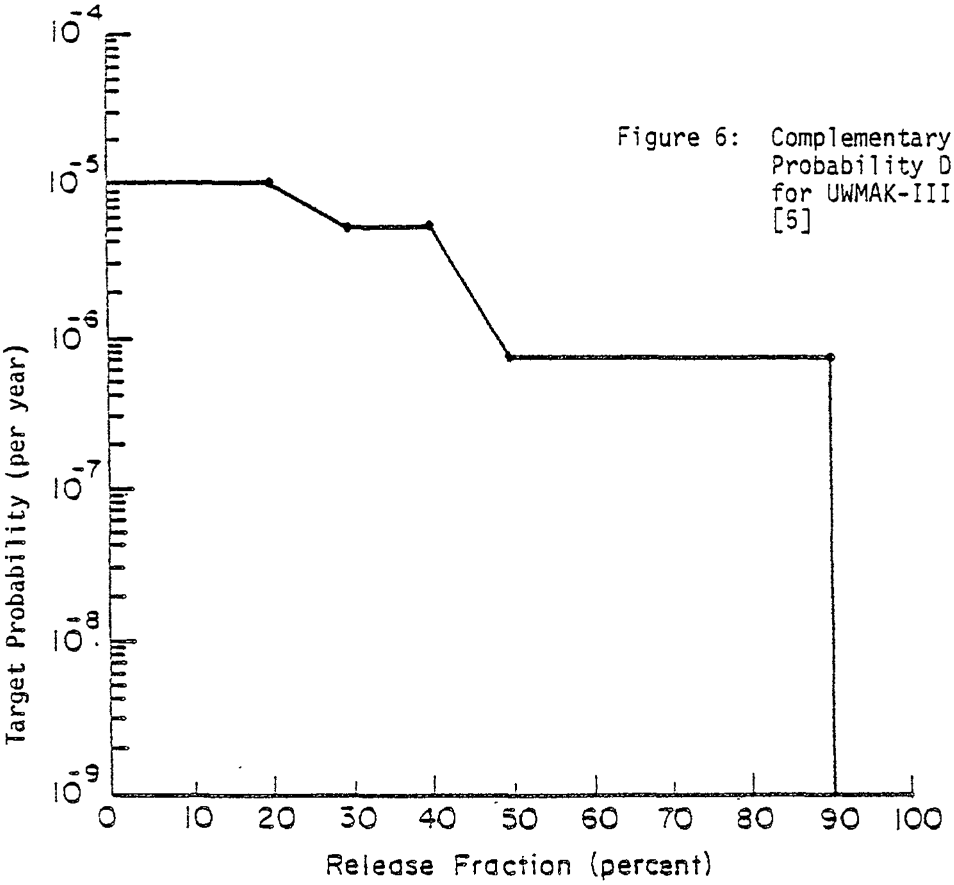

Figure 6: Complementary Maximum Tolerable Probability Distribution Function for UWMAK-III First Wall Releases [5]

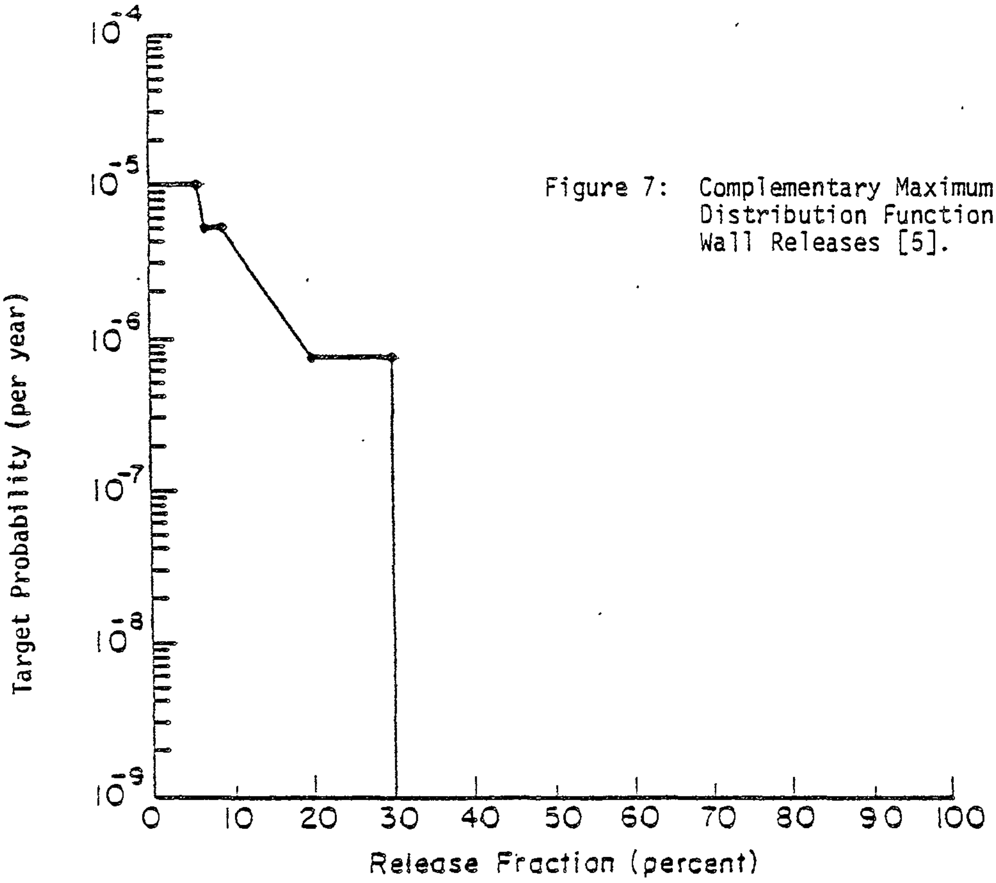

Figure 7: Complementary Maximum Tolerable Probability Distribution Function for UWMAK-I First Wall Releases [5].

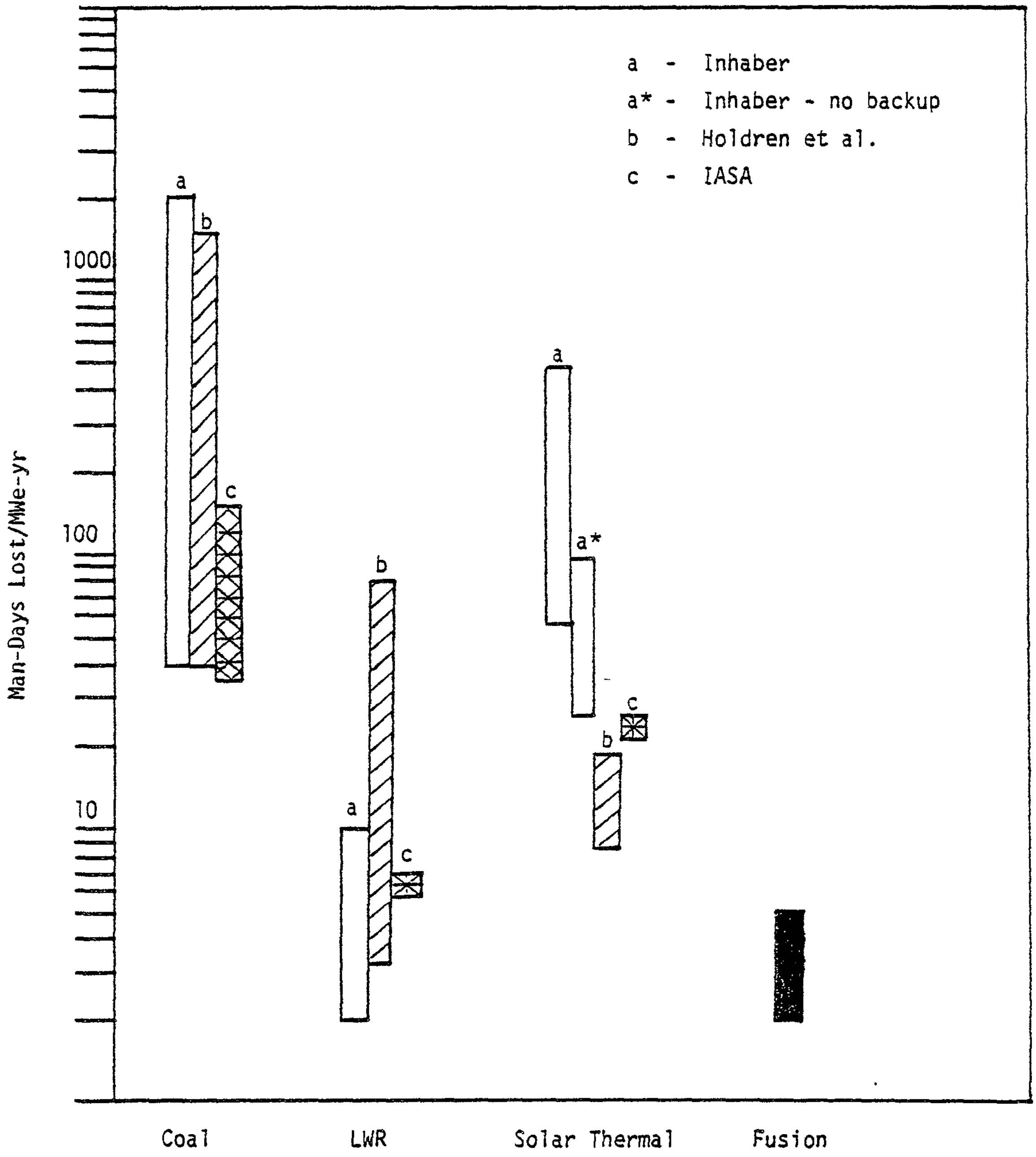

Figure 8: Total Power Cycle Risk Assessment.

DISCUSSION

J.GRUBER: We see that there is no linear relationship between the number of fatalities and the percentage of volatilization of the first wall. Is it easy to explain why this comes out of the model?

M.S.KAZIMI: Yes. If one looks at the dose-response curve there is a certain probability for fatalities associated with a certain dose level, e.g. a dose of 5 Sv (500 rem) is associated with 50% probability of fatality and hence the scale is not linear as to the response in terms of fatalities to the dose level itself. If one is adding just the dose level in rems it would be linear but the illnesses and fatalities are not linearly related to total manrems. This is based on the reactor safety study WASH-1400.

J.C.CHICKEN: The disadvantage of this lies in the absence of well defined acceptable risk criteria as well the large effect of public acceptance factors on the accepted risk. I thought perhaps there was a measure of agreement about what is, in quantified terms, the acceptable level of risk. Most safety people would say that a risk of one in a million was an acceptable risk and on this scale, when you get down to one in a thousand, that risk would be unacceptable.

M.S.KAZIMI: I do not know what the situation is in the UK, I am afraid, but in the USA the studies have shown that acceptability is dependent upon several factors: e.g. an immediate result, say a fatality, would have a different acceptable probability than a latent one; e.g. the acceptability of a new source of risk. A risk that has not been experienced for tens of years seems to be substantially lower than the acceptability of a source of risk that people have become used to. And therefore, when one defines the 10^{-6} as an acceptable level of risk, I think that this is a proposed acceptable level. There may be other factors that have to be taken into consideration as well. There does not seem to be, from studies I have seen, a simple criterion independent on other factors I have given you examples of.

J.C.CHICKEN: To some extent yes, I agree with you. Although the designer really has to have some target to aim at and for reactors, CEGB has given their designers firm criteria. Such criteria give the inspectorate a base to quantify the risk. Surely we need some kind of target for fusion reactors. That is really what you were saying. It must be defined by the licensing authority so that the designers know, in advance, what they are aiming at. Without such criteria one can waste a lot of time.

M.S.KAZIMI: Yes. I am not proposing any particular target but I am suggesting that we should try to look into these risk factors and develop a criterion by the time we will need one for an experimental power plant. And the USA Fusion Safety Programme Plan includes safety of design. But this work is still developing and progressing slowly in this direction.

J.SCHNEIDER: I have a question regarding the methodology of the mechanistical approach you were talking about. What is the relation of this mechanistical approach to full probabilistic risk analysis and how far can it be used for the establishment of event trees in probabilistic risk analysis? How sophisticated is your mechanistical approach at the moment? Did you discuss this with designers or even operators of facilities?

M.S.KAZIMI: As regards the first part of your question, obviously one needs to go through a deterministic analysis to be able to anticipate events that would then fall on the fault tree, i.e. a probabilistic risk assessment has to involve a certain degree of deterministic consequence evaluation and one cannot do the probabilistic one without the deterministic one. On the other hand one could do the deterministic analysis without necessarily combining them in a probabilistic fashion as in the earlier days of fission reactors.

With respect to the second part of your question we are very tight in with the designers of various reactors. There are actually two designers that I should talk about. Those that are involved in conceptual designs. The recent design of Argonne, the start for our design is one example. We have been quite close to the developments there and those that are actually designing reactor facilities, e.g. the TFTR. We have been talking to the TFTR people in terms of exchange of information but we have not applied any of our models to a particular facility yet.

J.SCHNEIDER: I think that the development of the accident sequences is a crucial point in the safety analysis.

M.S.KAZIMI: In the other parts of the programme there are specific targets which will address that e.g. in the TSTA facility there is a very specific task that is associated with the development of fault trees to be able to estimate risk and the probability of the various consequences.

J.L.ROUYER: Concerning volatilization of the first wall: You made a maximum accident calculation, which is probably not realistic, and you made a more credible deterministic approach in calculating that the heat is shared by all components. The figure you obtain is probably also not realistic. The sequence of the accident is not very well understood. One can make such type of analysis, probabilistic or maximum credible accident, when one knows much more about the design. But to compare design concepts between plant, I think one needs something quantified (as possible) but it cannot be a detailed probabilistic analysis.

M.S.KAZIMI: You have a valid point there. I have tried, at the very beginning, to caution you that we are dealing with paper designs and therefore the methodology is not meant to produce hard numbers for risk but rather guide-lines for comparison among various designs and various conditions. Really the objective of the study so far is more in terms of producing design guidelines than assessment of hazardness.

TRANSIENT ANALYSIS FOR FUSION BLANKET POWER ACCIDENTS

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Abstract

For a blanket using stagnant lithium as the breeder, helium as the coolant and a stainless steel first wall, transient temperature and thermal stress analyses have been performed to study the potential safety problem of such blanket modules.

1. INTRODUCTION

In the frame work of a safety study for a fusion power demonstration reactor, FINTOR-D [1], transient analyses for several types of blanket power accidents have been performed. For the blanket modules, with stagnant lithium as the breeder, helium as the coolant, and a stainless steel first wall and structual material, the following heat removal disturbances have been studied:

- helium depressurization due to a break of the outlet tube or the outlet plenum (= loss of coolant accident = LOCA);
- complete loss of helium coolant flow due to a sudden blockage of the inlet or outlet tube (= loss of flow accident = LOFA);
- power overshoot due to a temperature or density increase in the plasma at nominal coolant flow;
- plasma disruption onto the first wall.

Although the latter event is commonly not considered an accident, the response of the first wall to such a rapid process must be determined.

Transient non-linear heat-transfer calculations have been carried out, and the resulting temperature and thermal stress reponses of the first wall during the different events will be discussed in this paper. Both a blanket module without and with a radiation panel in front of the module has been examined. A radiation panel of welded stainless steel tubes, cooled by helium is considered.

2. THE REFERENCE BLANKET MODULE

The present safety analysis was a contribution to the conceptual design of the fusion power demonstration reactor FINTOR-D [1]. The blanket modules in that design are thin walled cylindrical vessels containing nearly stagnant liquid lithium as the breeder. The modules are arranged in rows around the plasma. The diameter of the modules is 52 cm and the length ~ 2 m. The thickness of the stainless steel vessel wall is 5 mm. High pressure (50 bar) helium gas flowing through cooling tubes was chosen as the primary coolant for the modules. The coolant tubes have the shape of spirals and are arranged on parallel planes perpendicular to the axis of the cylindrical vessel, as illustrated in fig. 1. Each spiral-shaped tube is connected to an inlet- and an outlet manifold which are also situated within the modules. The length of each tube is 5 m and the inner- and outer diameters are 12 mm and 14 mm respectively. The distance between the coolant tubes (the pitch) is 22.7 mm. The helium gas, entering the module with a temperature of 150°C first passes closely the thermally high-loaded first wall and then it is heated up moving radially inward through the module to leave the module with an outlet temperature of 350°C. For the reference average neutron wall loading of 1.34 MW/m^2, the power production within that part of the first wall facing the plasma was calculated to be 15 W/cm^3. It was assumed to decrease to 3 W/cm^3 at the rear side of the module.

The surface radiation heating on the first wall was calculated to have a maximum value of 40 W/cm^2 at the front side of the modules. It was assumed to decrease to zero at both mid sides of the modules. Per unit length of module a total heat of 0.70 MW has to be removed by the helium gas flow. This requires a helium gas flow velocity of 30 m/s through the tubes, with a resulting steady state heat transfer coefficient of 2340 W/m^2 °C.

3. THE NUMERICAL MODEL FOR THE CALCULATIONS

The finite-element code MARC [2] has been used to find the initial steady state temperature distribution and the thermodynamic response of the blanket module to coolant flow disturbances or power load variations. The complicated geometry of the modules necessitated certain simplifications in the numerical model to avoid too excessive computing time for the transient analysis. Consider the r-z plane through the axis of the cylindrical module. This plane is perpendicular to the first wall and to the planes of the spirally shaped coolant tubes. To simplify the real situation to a representative 2-dimensional substitute, only a strip in the r-z plane between the front side of the first wall and the outlet tube is considered, and straight coolant tubes perpendicular to the r-z plane are assumed. The layout of this region is given in fig. 2 and consists of 24 two-dimensional 8-node elements and 123 nodal points in total. The first wall is represented by element 1. The other elements represent the liquid lithium.
The lower and upper boundaries of this region are two symmetry planes, one through the axis of the helium tubes of one spiral, the other just between two adjacent spirals. Therefore, the thickness of the strip as shown is 11.35 mm and is equal to half the pitch between the spiral tubes.
The steady state temperature of the helium in the successive tube passages was calculated by hand, taking into account the volumetric power density in the lithium when the helium is heated up gradually in the spiral from its inlet to its outlet temperature.
In the calculations the surface heat fluxes, the power densities, and the heat transfer coefficients at the boundaries are time dependent input parameters. The temperature dependent material properties of stainless steel and liquid lithium are taken into account. The liquid lithium is assumed to be stagnant, so convective heat in the lithium is not considered. The assumed power density in the lithium and the calculated steady state temperature profile are also given in fig. 2.

4. HELIUM DEPRESSURIZATION DUE TO A PIPE BREAK

In case of a pipe break in the helium system the blow-down and depressurization of the helium gas can be evaluated [3]. Assuming isothermal depressurization the depressurization of the helium is given by

$$p(t) = p_0 \exp(-\frac{t}{\tau^*}). \tag{1}$$

In this simple expression the depressurization time constant τ^* is given by

$$\tau^* = \frac{V_0}{A_0} \left[RT\kappa \left(\frac{2}{\kappa+1}\right)^{\frac{\kappa+1}{\kappa-1}} \right]^{-\frac{1}{2}}, \tag{2}$$

where V_0 = helium gas volume, A_0 = break area, and $\kappa = c_p/c_v$.
For helium, the time constant will be approximately:

$$\tau^* = 0.0304 \frac{V_0}{A_0 \sqrt{T}}, \tag{3}$$

where τ^* is expressed in s, V_0 in m^3, A_0 in m^2, and T in °K.
Assuming adiabatic depressurization the pressure is given by

$$p(t) = p_0 \left[1 + \left(\frac{\kappa-1}{2}\right) \frac{t}{\tau^*} \right]^{\frac{-2\kappa}{\kappa-1}}, \tag{4}$$

and for helium $p(t) = p_0 (1 + \frac{t}{3\tau^*})^{-5}$. For $t < 5\tau^*$ adiabatic depressurization will be faster than isothermal depressurization, see fig. 3.
The heat transfer coefficient of the helium gas flow decreases during the depressurization according to the Dittus-Boelter equation $h \sim (\rho v)^{0.8}$.

For the isothermal depressurization this results in $h(t) = h_0 \exp(-0.8t/\tau^*)$, and for adiabatic depressurization $h(t) = h_0 [1+t/3\tau^*]^{-2.4}$. From fig. 3 it is seen that the heat transfer coefficient decreases much faster for the isothermal approximation and the latter is chosen in subsequent calculations.

For a pipe break inside the inner vacuum containment (~ 5000 m^3) the helium blow-down causes a pressure build-up in the plasma vessel up to about 2 bar assuming that the escaped hot helium is subsequently cooled down to room temperature. Helium with an initial pressure of 50 bar will reach this equilibrium situation in about $4\tau^*$. The resulting plasma contamination will stop and shut down the fusion power production automatically much earlier. A pipe break in the secondary containment (~ 150.000 m^3) gives a residual pressure << 1 bar and the blow-down period will extend over more than $5\tau^*$. A detection system then must be provided for safe shutdown of the fusion process. The transient analysis therefore has to extend over a period greater than $5\tau^*$. A pipe break in the outlet tube outside one blanket module, will be taken as a representative loss of coolant accident (LOCA). Such a break will have an area of about 0.01 m^2.

For a helium volume of 400 m^3 the corresponding depressurization time constant then will be $\tau^* = 50$ s. Compared to the total circulating mass flow with a circulation period in the primary coolant loop of about 2 s, this reference LOCA is a relatively slow proceeding event.

A pipe break in one of the main helium feeding or collecting headers outside the reactor (with diameters of the order of 1 m) causes rapid depressurization with a time constant of $\tau^* \sim 1$ s. Such an event can be supposed as being as severe as the worst loss of flow accident (LOFA), where the helium flow through a blanket module is suddenly stopped by a coolant tube blockage (for the calculations this is equivalent with $\tau^* = 0$).

5. RESULTS OF TRANSIENT CALCULATIONS FOR THE BLANKET MODULES WITHOUT RADIATION PANEL

For the initial steady state condition (see fig. 2) the maximum temperature in the first wall is 570°C and the temperature drop over the first wall, ΔT_w, is 107°C. The time in a start-up sequence to reach this steady state temperature profile is calculated to be about 30 s.

5.1. Helium coolant disturbances (LOCA and LOFA)

The transient behaviour of the first wall and the lithium in a LOCA with a small depressurization rate ($\tau^* = 50$ s) and with a larger depressurization rate ($\tau^* = 25$ s), both with continuing power production, is shown in fig. 4. The temperature of the front side and the rear side of the first wall, as well as the minimum lithium temperature are given. For the two LOCA transients a maximum first-wall temperature of 900°C will be reached within 160 s and 110 s, respectively. The temperature drop ΔT_w over the first wall decreases during the transient from 107°C to about 80°C for both cases. Direct or even delayed action of the safety system, initiating a rapid shutdown of the power production, results in a safe course of the depressurization event, as illustrated also in fig. 4. The temperature transient for the LOFA, i.e. complete helium flow blockage, is given in fig. 5. With continuing full power production the front side of the first wall will reach a temperature of 900°C within 50 s and will melt after about 150 s. Shutdown of the power production process within 50 s results in a safe decrease of the wall temperature until there is a slightly increasing temperature effect with a rate of $\doteq 0.1$°C/s due to the afterheat production. The temperature drop over the first wall in the LOFA will also decrease to 80°C at the time where the 900°C limit is reached.

The thermal stress at the surface of the first wall is given by

$$\sigma_{th} = \pm \frac{\alpha E}{2(1-\nu)} \Delta T_w, \tag{5}$$

where E = Young modulus, ν = Poisson ration, and α = linear expansion coefficient.

The thermal stress relative to the elastic yield strength σ_y can be given by

$$\frac{\sigma_{th}}{\sigma_y} = \frac{\alpha E}{2(1-\nu)\lambda\sigma_y} \cdot d_w q'', \tag{6}$$

where q'' = heat load on the first wall of thickness d_w (with $\Delta T_w = \frac{q'' d_w}{\lambda}$).

The material properties σ_y and the ultimate tensile strength σ_u are strongly dependent on temperature as are shown in fig. 6 for unirradiated 316 SS.

The thermal stress parameters $\frac{\alpha E}{2(1-\nu)}$ and $\frac{\alpha E}{2(1-\nu)\lambda\sigma_y}$ vary only slightly in the temperature range of 200-900°C, as is shown in the figure. For a temperature drop of 80 - 100°C and for a wall temperature above 600°C, the first wall will be in the plastic deformation domain.

This means that for a proper analysis of wall behaviour a more complicated elastic-plastic-strain calculation has to be performed. Assuming that no plastic deformation of the wall occurs during the LOCA and LOFA events, the thermal stress exceeds the σ_u limit above 900°C. Therefore, it is plausible to take this temperature of 900°C as a possible upper limit, below which proper shutdown might avoid wall failure.

5.2. Over-power transient

Due to plasma instabilities, disturbances in fuel supply, or to temperature control, the power production in the plasma might fluctuate. The transient behaviour of the blanket module with nominal helium cooling to a sudden over-power production is given in fig. 7. An over-power of 125% results in a maximum wall temperature of 650°C and a temperature drop over the first wall of 128°C. The resulting thermal stress will be below 1.5 σ_y. 150% over-power leads to a maximum temperature of about 900°C within 60 s. A temperature drop over the first wall of 180°C will already be reached within 10 s, causing very excessive thermal stresses > σ_u with a wall temperature still below 700°C (see fig. 6).

5.3. Plasma disruptions

A plasma thermal energy of 400 MJ and a magnetic energy in the plasma of also 400 MJ are assumed in the calculations. Recent calculations [4,5] show that plasma disruption times of about 25 ms may be expected, which are longer than earlier estimates of about 1 ms. A uniform plasma disruption in 25 ms onto a total first wall surface of ~ 1000 m^2 results in an average surface heat flux of 16 MW/m^2. By lack of sufficient knowledge of the disruption process, a constant heat flux on the first wall during the disruption time is compared with a time-varying heat flux according to the plasma expansion model of ORNL [4]. The resulting temperature response of the first wall is given in fig. 8. In ref. [4] it is expected that about 30% of the magnetic energy will be dissipated as Joule heating in the first wall during the disruption, the rest of the magnetic energy will be dissipated in other structures.

A uniform and constant dump of this fraction of the magnetic energy in the first wall results in a volumetric heat production of 1200 MW/m^3 (1.2 kW/cm^3) during the disruption.

For the case of constant heat flux, a uniform dump of the plasma energy together with 30% of the magnetic energy on all modules is compared with a uniform dump of only the plasma energy on all the modules. The results of these calculations are also given in fig. 8. The maximum temperature rise of the front side of the first wall will be 285, 291 and 310°C for the 3 different cases 1, 2 and 3, respectively, as depicted in fig. 8. The effect of the volumetric Joule heating is negligible compared to that of the surface heat flux.

A uniform plasma disruption on all modules causes a rapid temperature increase of the front side of the wall to about 860°C. The rear side of the wall remains at a nearly constant value in accordance with the long thermal diffusion time constant for the stainless steel wall of $\tau_{diff} = \frac{d_w^2}{a} = 5$ s (d_w = wall thickness, $a = \frac{\lambda}{\rho c}$ = thermal diffusivity).

For the rapid plasma disruptions separate MARC-calculations for the first wall only, with fine mesh distances, have been carried out to obtain proper numerical results. The temperature dependent material properties are taken into account. If temperature independent material properties are assumed then the temperature response of the first wall to a sudden heat flux q'' may also be approximated analytically [6]. For plasma disruption times short compared to the thermal diffusion time of the wall the temperature of the front side of the wall then increases with

$$\Delta T(t) = \frac{q''}{\lambda} \sqrt{\frac{4\,at}{\pi}} \,. \tag{7}$$

For the plasma disruption as considered this will be 290°C at the end of the disruption. This is slightly higher than found numerically in the case of temperature dependent material properties. Therefore, expression (7) can be used to find the thermal response for varying wall loads or varying disruption times. Melting of the surface of the wall then may be expected for heat fluxes > 50 MW/m^2.

The peak thermal stress in the heated wall surface due to the sudden increase ΔT^{pulse} of the surface temperature, is $\sigma_{th}^{pulse} = \frac{\alpha E}{(1-\nu)} \Delta T^{pulse}$ and is twice as high as the thermal stress due to a slow-varying or steady temperature drop of the same magnitude. Therefore, even the uniform plasma disruption will cause thermal stresses which may exceed the tensile strength limit with consequentially a high probability of gross wall failure. Equation (7) shows that only very long disruption times apparently may reduce the failure probability.

6. THE BLANKET MODULE WITH RADIATION PANEL

The main argument for the incorporation of a radiation panel in front of the first wall is to mitigate the thermal stress in the first wall which is dominated by the surface heat load. The radiation panel is cooled to remove the radiation heat flux caught by the radiation panel. The remaining part of the blanket module now removes less heat.

The steady state temperature profile and the transient behaviour for such a blanket module design shows that this design is very attractive. During a LOFA, see fig. 9, the temperature of the wall of the module now increases from a steady state temperature of 280°C to 900°C within 250 s. The temperature drop over the first wall (in steady state $\Delta T_w = 10^{\circ}C$) will decrease to about 2°C. Below the melting point (reached after about 400 s) no failure of the blanket module is expected. Therefore, the safety analysis now has to be focussed on the transient behaviour of the radiation panel.

A radiation panel of a closed layer of welded helium-cooled circular tubes of stainless steel is considered. The inner and outer diameter of the tubes are 10 and 12 mm, respectively. Helium with the same conditions as in the case of the reference blanket module (p = 50 bar, inlet temperature 150°C, outlet temperature 350°C), and with a coolant flow velocity of 50 m/s through the tubes, removes the radiation heat flux.

Only the side of the tube facing the plasma, is subjected to a heat flux. The rear side of the tube facing the blanket module, is free from a heat influx. This asymmetric heat flux results in a non-uniform circumferential wall temperature, see fig. 10.

A heat flux perpendicular to the surface of the tubes, decreasing according to $q'' = q_0'' \cos\phi$ from a maximum of 0.40 MW/m^2 at the mid front side to zero at both flanks of the tubes, where the tubes touch each other, is assumed. The maximum surface temperature will be 374°C and the maximum temperature drop over tube wall (having a thickness of 1 mm) is 22°C. In the case of a uniform heat flux the values will be 390°C and 24°C, respectively.

In a start-up sequence this steady state temperature profile will be reacned within 6 s.

In case of a helium flow blockage with continuing fusion power production the temperature of the tube increases rapidly, as is shown in fig. 11. Within 10 s the front side of the tube will reach the earlier chosen 900°C limit and melting occurs within 30 s. This is a much faster transient than the transient in the unprotected blanket module for the comparable flow blockage. The temperature drop over the wall thickness, however, reamins small ($\Delta T = 8^{\circ}C$ at 900°C), so the thermal stress in the tube related to this temperature drop will be far below the yield strength (see fig. 6).

A more serious thermal stress problem is presented by the non-uniform circular distribution of the heat flux. 9 s after the start of the LOFA the temperature of the outer surface of the tube varies from 900°C at the front side to 350°C at the rear side, as is also illustrated in fig. 10.

The thermal response of the tube to a uniform plasma disruption in 25 ms is given in fig. 12. The front side of the tube wall will rise with 315°C. The inner surface of the tube wall remains nearly constant, partly due to the forced convective cooling (the thermal diffusion time constant of this wall with a thickness of 1 mm is 0.24 s).

A total temperature drop over the wall thickness of 325°C is expected at the end of the plasma disruption. A temperature gradient of the same magnitude is found in circumferenital direction in a small region at the flanks where the tubes touch each other. This is illustrated in fig. 10. These radial and circumferential temperature drops lead to high thermal stresses which may exceed the tensile strength limit. Recently [5], a local deposition of the main part of the plasma energy on a smaller part of the first wall area was quantified in some extent. Local heat fluxes up to 5 to 10 times the uniform average heat flux may be expected, resulting in partial melting of the first wall surface. For the radiation panel as considered, a localized plasma disruption with 5 x 16 MW/m^2 and 10 x 16 MW/m^2 respectively is analysed. The temperature responses are also given in fig. 12. Local melting to a depth of 0.04 mm and 0.3 mm, respectively, will occur.

7. CONCLUSION

With a radiation panel in front of the blanket modules the safety of the blanket modules is better guaranteed. The failure probability of (single tubes of) the radiation panel, might be higher than those of the blanket module without radiation panel. However, the consequences of such a failure are probably less severe. A failure of the wall of the blanket module leads to an expulsion of liquid lithium in the plasma chamber and this will certainly stop the plasma reaction, but may possibly also contaminate other systems such as the vacuum pumps, and propagate other accidents such as a lithium fire. Failure of tubes of the radiation panel results in a helium expulsion, which increases the helium inventory of the plasma and stops the process. A hazardous consequence is the build-up of the pressure which the vacuum vessel has to withstand.

The calculations show that coolant flow disturbances result in temperatures and thermal stresses in the first walls, which are comparable to or even lower than those resulting from plasma disruptions. Only very long disruption times or the application of first wall coatings may probably reduce the failure probability.

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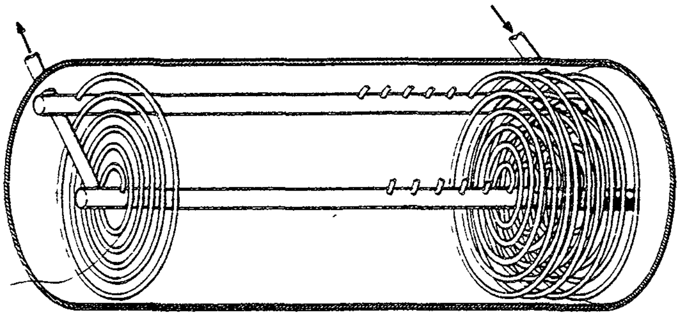


Fig. 1. Blanket heat transfer system with spiral-wound cooling tubes

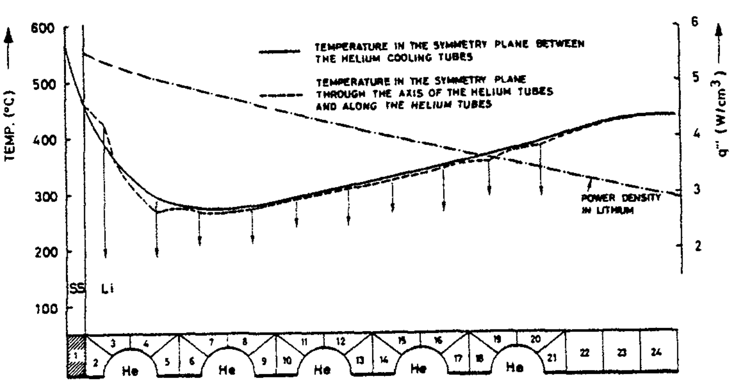


Fig. 2. STEADY STATE TEMPERATURE DISTRIBUTION IN BLANKET MODULE

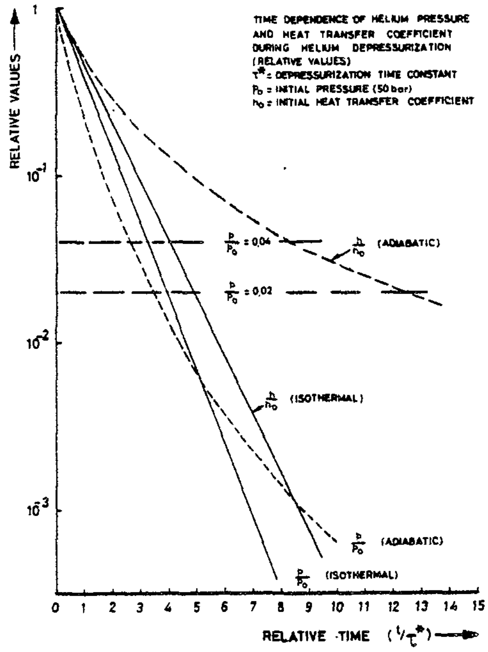


Fig. 3. HELIUM DEPRESSURIZATION

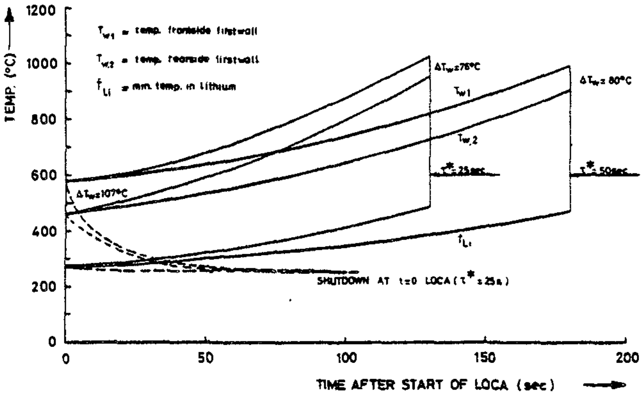
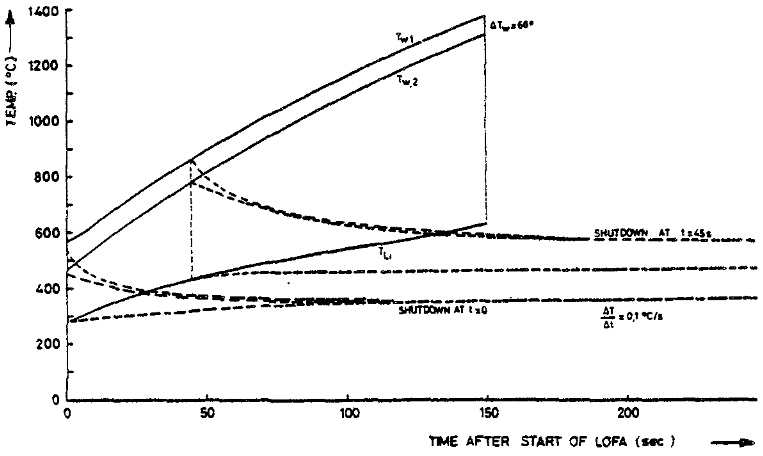


Fig. 4. TEMPERATURE TRANSIENT IN FIRST WALL DURING A LOCA (HELIUM DEPRESSURIZATION WITH CONTINUING FULL POWER)



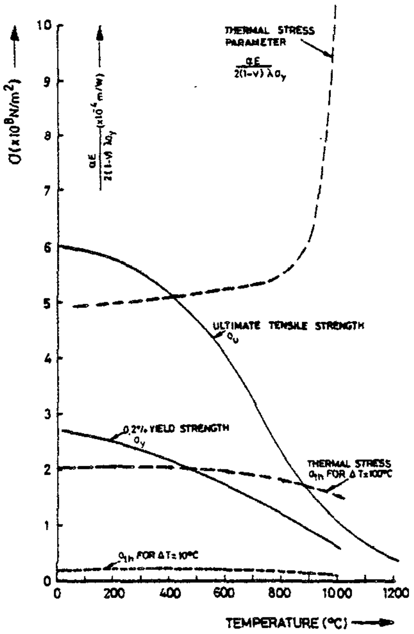


Fig. 5. TEMPERATURE TRANSIENT FIRST WALL DURING LOFA (HELIUM BLOCKAGE WITH CONTINUING POWER)

Fig. 6. UNIRRADIATED 316 SS THERMAL STRESSES AND STRENGTHS

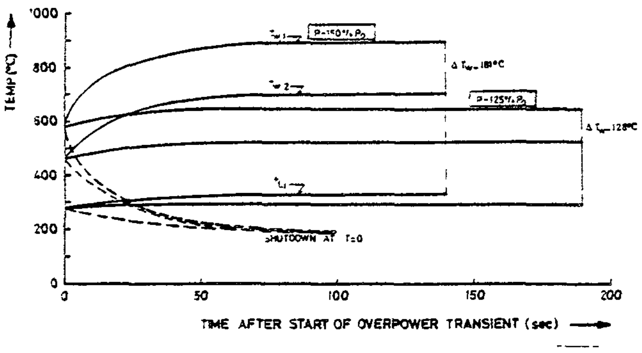


Fig. 7. *TEMPERATURE TRANSIENT FIRST WALL DURING OVERPOWER*

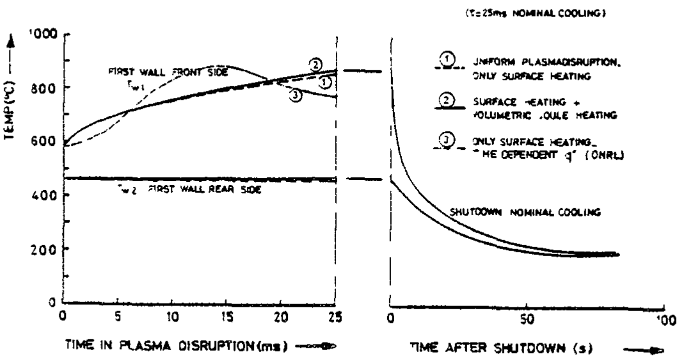


Fig. 8. *FIRST WALL TEMPERATURE DURING PLASMA DISRUPTION*

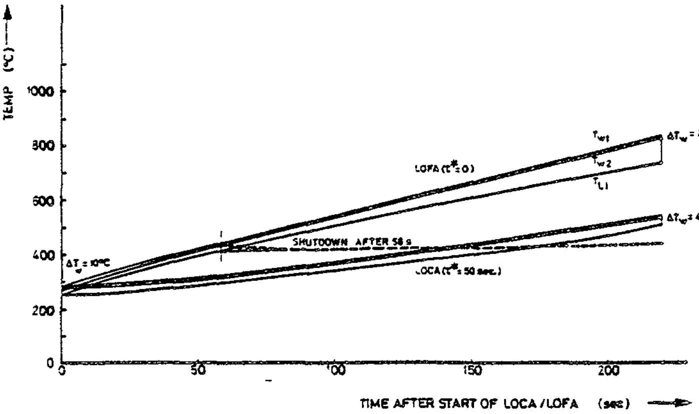


Fig. 9. *TEMPERATURE TRANSIENT FIRST WALL DURING LOCA/LOFA FOR THE BLANKET MODULE WITH RADIATION PANEL*

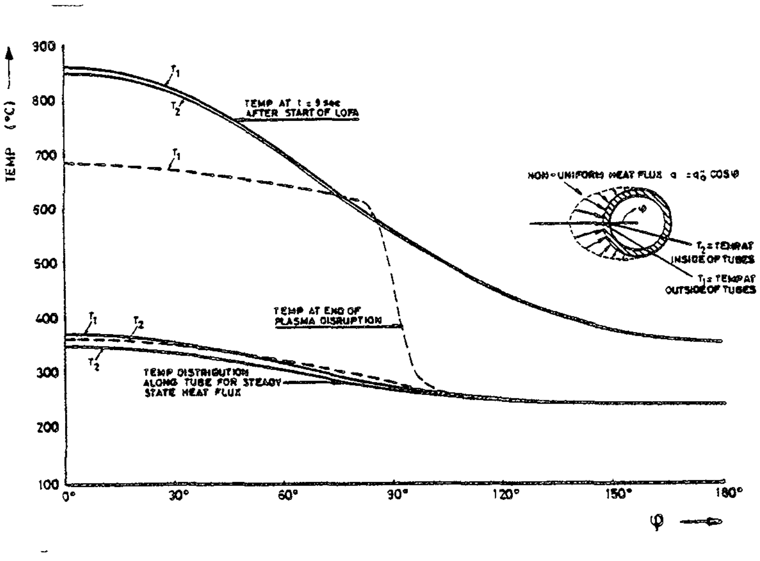


Fig. 10. *RADIATION PANEL: CIRCUMFERENTIAL TEMPERATURE DISTRIBUTION ALONG TUBE*

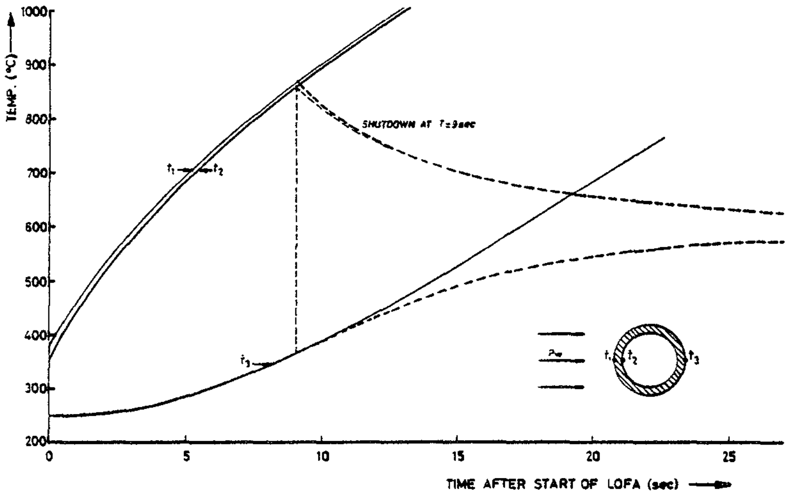


Fig. 11. TEMPERTURE TRANSIENT OF TUBES OF RADIATION PANEL DURING LOFA WITH CONTINUING POWER PRODUCTION

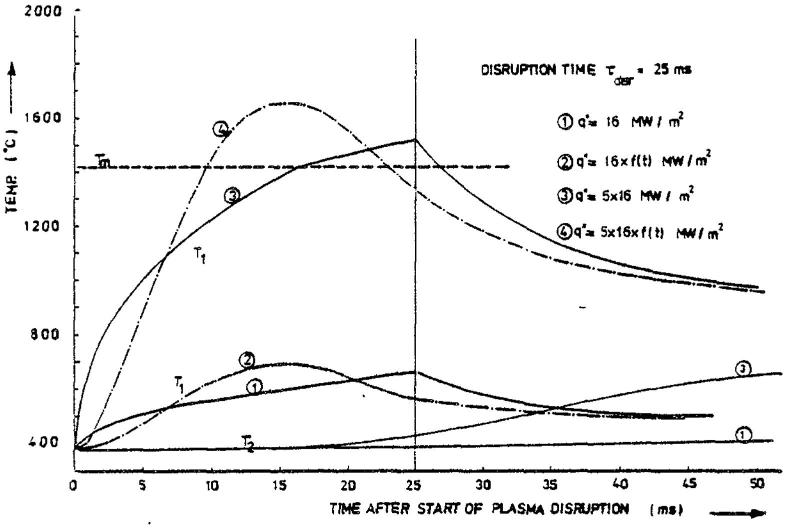


Fig. 12. PLASMA DISRUPTION ON RADIATION PANEL

DISCUSSION

J.G.CROCKER: You showed the temperature profile of the radiation panel at 9 sec. What do you have for that maximum temperature as a function of time? What is the heat-up rate at the end of your calculation? That temperature should be rising at the end of your calculation.

H.T.KLIPPEL: My calculations ended at 50 sec. There was no significant rise of temperature (for the blanket itself 0.1 °C/sec).

M.S.KAZIMI: Is this instantenous loss of flow?

H.T.KLIPPEL: This loss of flow is an instantenous loss of flow where you have several blokades in the outlet. It is a kind of adiabatic situation.

M.S.KAZIMI: All the tubes?

H.T.KLIPPEL: No, of one blanket module with one inlet connection and one outlet connection. There is no redundancy in connection of coolants to blankets. There is a module of 2 m times 1/2 m and here all the tubes have an inlet and one outlet connection. In my calculations I supposed a blokade of the coolant tubes, and a loss of coolant accident respectively. In the latter I supposed a break of these tubes. An outer tube break results in a time constant of about 50 sec and a break of the inner tube gives a time constant of about 25 sec.

M.S.KAZIMI: What is the material surrounding the cooling tubes?

H.T.KLIPPEL: Also stainless steel, 1 mm thick stainless steel tubes with an outer diameter of 12 mm.

M.S.KAZIMI: I am not quite clear. You have the shell for which you calculated the temperatures and you have the tube for the coolant going through that shell. What is in between the module shell and the tube? Is there breeding material?

H.T.KLIPPEL: I do not quite understand your question. We have here liquid lithium in the tank. The heat transfer through the liquid lithium is only put as conductive heating, not convective heating. There must be a slow movement of the liquid lithium for the tritium reprocessing. But that flow is skipped in this calculation. The calculations can also be repeated for all kinds of blankets with helium coolant in pipes. My intention last year was to do such calculations in a parametric way, to look at higher wall loads, but the calculations changed to plasma disruptions. It was also the intention to do some alternative blanket designs e.g. solid lithium and water coolant. For the European INTOR team I have recently started to look at the same aspect I have discussed here, at the loss of coolant, loss of flow accident, but unfortunately for the new INTOR design, with water cooling, there are more troubles in the calculations than in helium depressurization.

M.S.KAZIMI: What do you think would be the impact of accounting in the third direction of your calculation? Since this is a two dimensional calculation.

H.T.KLIPPEL: Yes, this calculation is a bit simplified but we have carried out a three dimensional calculation for the steady state situation. Three dimensional calculations for this vessel were very difficult but the situation in that calculation was that one has a temperature profile which remains at about the same level over a wide angle of the plasma phasing situation because the design is such that the distance between helium tube and vessel will be larger for larger angle and that means that there is lower heat transfer from the wall to the coolant. This in turn means that over a wide angle of about 54° the temperature profile over this first wall surface is about the same. I looked at the most severe conditions in this section of the blanket module.

J.G.CROCKER: We briefly have done some analysis similar to this on one of our INTOR designs and we have essentially come-up with the same conclusions. The conclusions in effect are that there is plenty of time for either a loss of coolant or loss of flow to take corrective action, if necessary. Actually we do have a calculational technique that has a shut-down mechanism of evolution of impurities of the first wall which shows that even if one does not take corrective action the plasma will terminate in sufficient time. But the main conclusion is that one has plenty of time to take corrective action to shut off the plasma in either one of these events. This was for a different design, for a different coolant, but for the same first wall and essentially the same wall loading (1.3 MW/m^2).

H.T.KLIPPEL: For INTOR, for a stainless steel wall, I have made a rough calculation. In the January 1981 Meeting of the INTOR Workshop at Vienna it was decided to take a first wall of about 14 mm thickness, cooled by water (water coolant pressure is 10 bar) with a maximum temperature of 100°C. The maximum temperture of the first wall is fixed at 350°C. The temperature drop over the first wall was about 130°C (for 14 mm). That meant that the temperature of the rear side of the first wall was, with 120°C, a little bit higher than 100°C. The calculations were done for a complete blockade for the water cooling heat transfer. This is a most pessimistic situation, an adiabatic heating-up. The first wall reached the melting point after about 200 sec. My calculations showed that, within a very small time scale, the original temperature drop of about 130°C goes down to about 50°C and remains constant during the rest of the excursion. That means that the temperature drop over the first wall gives not much problems in terms of thermostresses. The melting point limit is the most decisive one and it takes about 200 sec to reach that point. The most severe conditions for such accidents in INTOR is the melting point of lead (327°C). If one looks at the adiabatic situation (the worst condition for the lead), the lead will melt in about 60 sec. These 200 sec for the first wall in INTOR have to be compared with my calculations for the same tubes.

J.G.CROCKER: What we showed in our calculations analysis, it was a different first wall design, was that we reach first wall melting in 300 sec. For any situation, whether one turns off the plasma or not, one always reaches the melting point in lead. Lead, that is used in the neutron multiplier, is the crucial material.

H.T.KLIPPEL: In the worst conditions one has about 1 Minute of time.

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SUPERCONDUCTING MAGNET SAFETY

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Abstract

One of the major components in a fusion reactor for which a safety analysis must be carried out is the magnet system.

Most of the possible disturbances influencing the operation of superconducting magnets lead only to a quench, defined as an "abnormal operating condition" which causes just a temporary shut down of the magnet system without a damage, if the system is well designed.

More unlikely are accidental events which are associated with the generation of high power arcs. In these cases, single current arcs, e.g. at broken current leads, will lead to moderate damage only, but with the necessity of a longer shut down period for repair or replacing. Severe damage can only occur if in a multiple current arcing, starting by broken conductors, a wide-spread rupture of the winding occurs and the final high power arc burns through the coil case damaging other coils and reactor components.

In a very hypothetical event the simultaneous rupture of the complete winding at two locations at least 1 m apart leads to missile generation due to the electromagnetic forces in the background field. The kinetic energy which the flying piece can get will be less than the values assumed for airplane crashes with the containment of modern fission power plants.

1. *Introduction*

It is well recognized, that superconducting magnets are indispensible for fusion reactors with magnetic confinement. Even today, several large scale experiments in operation or under construction such as T7, T15 |1|, Tore II |2| and MFTF-B |3| use superconducting magnets. Reliability and safety aspects are of course not neglected in designing these experiments. They will become more pronounced for even larger fusion devices, with probably ignited plasma, planned for the next decade, such as the discussed international "INTOR" |4| and national project plans. Ignition means use of tritium and activation of structural materials by the fusion neutrons. Therefore such a facility must be designed so that failures will not cause intolerable environmental impacts.

Thus, the magnet system is one of the major components for which a safety analysis must be carried out. In a recent paper we have tried to give an answer to the possible failure modes and the hazard potential of superconducting magnets, especially in fusion devices |5|. The presentation now will summarize some results of these considerations together with that of further investigations carried out very recently.

2. *Classification of magnet failures*

In fusion devices the magnets have a large volume, typically 1000 m^3, implying a large liquid helium content in sophisticated cryostats with high vacuum thermal insulation and inner and outer pressure vessels. Further, the magnetic field strength will be high, about 8-12 T so the forces to be supported by structure and the stored electromagnetic energy (~10-50 GJ) are very high, too.

For the safety analysis of the magnet system it is important to distinguish between "abnormal operating conditions", which cause a temporary shutdown of the magnet system without a damage, and "accident situations" with the possibility of subsequent destruction. The best known magnet failure is the "Quench", a sudden transition of a part of the superconductor to normal conductivity. It is important to recognize that a quench is only an abnormal operation condition as demonstrated in |5| and can usually be well considered in the design so that no damage occurs.

3. *Abnormal operating conditions*

The above mentioned "Quench" is the only abnormal operating condition occurring: A normal conducting zone in the winding remains stationary or propagates without a probability for short-time recovery (Fig. 1). Stabilization of the conductor and the winding against quenching is one of the main design criteria in all superconducting magnets. Especially in large magnets as those for fusion devices, the safest, so called "cryogenic stability" is foreseen as design criterion. Cryogenic stability means that the heat associated with a local disturbance, even resulting in the creation and formation of a normal conducting zone in the superconducting wire, can be transfered to the cooling medium, so that the zone disappears again within a short time. For that purpose the wire

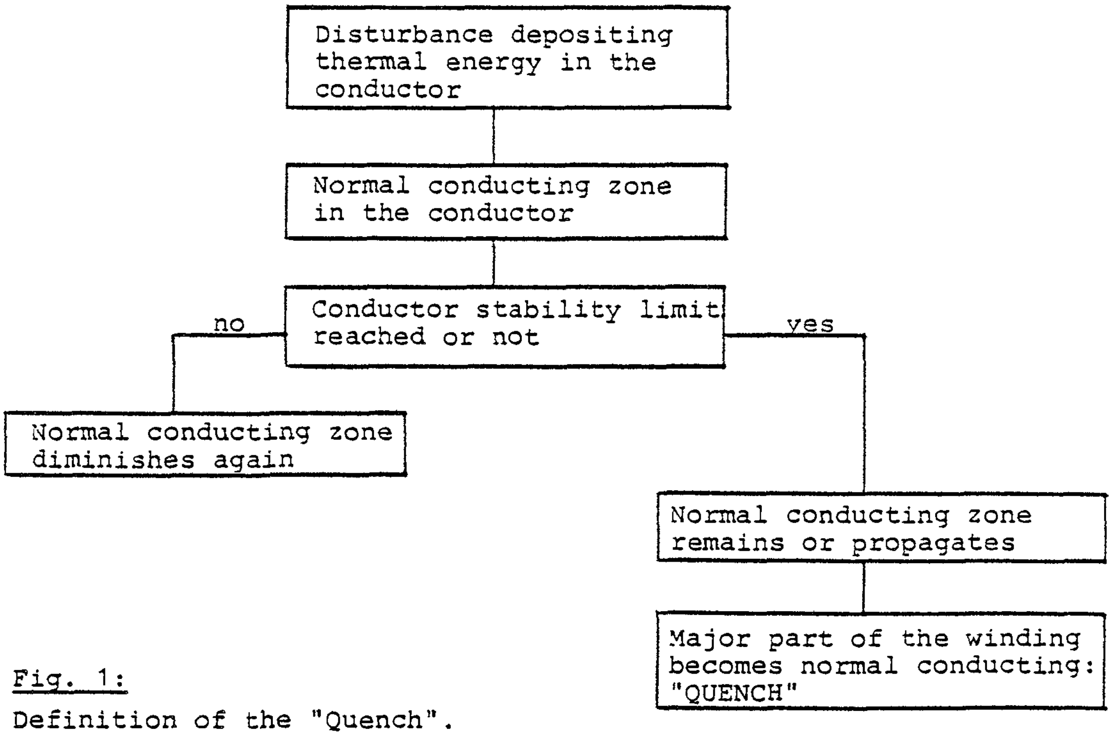

Fig. 1:
Definition of the "Quench".

has to be wetted, at least partially, by the cooling medium (He) for good heat transfer. The conditions for cryogenic stability are then fixed by the heat transfer to the helium (influenced by temperature, pressure, cooling channel dimensions, flow velocity, surface conditions, etc.), the maximal energy of possible disturbances, and the amount of stabilizing normal conducting material (e.g. Cu) in the conductor.

Therefore, a quench in a fusion magnet can only be triggered if the conditions for cryogenic stability are disturbed by the following events:

- Lack of cooling, caused by a sudden loss of helium or a blockage of cooling channels;
- Sudden local heating with absolute values far above the design limits, e.g. caused by a sudden movement of a sufficiently large part of the winding and a.c. loss heating of this part due to the field change;
- Lack of cooling, caused by evaporation of the helium in case of a vacuum breakdown.

As mentioned, all superconducting magnets must be built in such a manner, that together with their supply and discharge system they can take quenches without damage.

This can be done, because for all sources of failure initiating a quench the subsequent events finally end up in the following consequences (as demonstrated in detail in |5|):

- fast external safety discharge with
  - coil heating below 80 K,
  - a maximum discharge voltage given by the discharge circuit,
  - evaporation of the He in the coil with predictable pressure increase and He-blow-off through safety lines
  - redistribution and enhancement of asymmetric mechanical forces in a coil system
  - inductive current enhancement in neighbouring coils,

or, in case of malfunction of quench detection or the external discharge circuits

- pure internal discharge within the winding with the same consequences as above, but coil heating to temperatures up to about 300 K.

All these consequences can be considered appropriately in the coil design, as it will be indicated briefly below.

3.1 Coil heating

Fig. 2 gives the principle of a simple discharge circuit for a superconducting coil. The discharge is initiated by opening of the power switch between coil and power supply. In case of a quench this will be triggered by the quench detector monitoring the voltages in different coil parts.

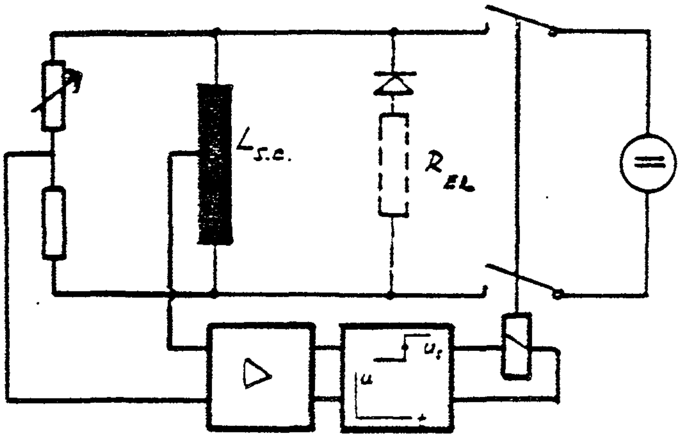

Fig. 2: Simple discharge circuit for a superconducting coil.

In case of a quench the current flowing in the coil creates Joule-heating in the conductor without further significant heat removal by the cooling medium. This process is governed by the heat balance equation together with the electric discharge equation. Often the simplified so called "hot spot criterion" can be used for an estimate. The simplification contains exponential current density decay j(t) with a constant discharge resistance and assumes the same specific heat for superconductor and stabilizer:

$$J \equiv \int_{4K}^{T_{max}} \frac{c_{St}(T)}{\rho_{St}(T)} \, dT = \frac{1+\alpha}{\alpha} \cdot j_o^2 \cdot \tau_n \qquad (1)$$

j_o is the overall current density in the conductor, τ_n the discharge time constant, α the stabilizer to superconductor ratio and $c_{St}^{(T)}$ and $\rho_{St}^{(T)}$ are the specific heat and electrical resistivity of the stabilizer respectively. The integral J is available for Cu, e.g. in tabulated form, so that the maximum conductor temperature T_{max} ("hot spot") can easily be determinated.

Within the design, the values for α, j_o and τ_n must be adjusted in accordance to this equation so that T_{max}<80 K. This value is chosen because up to 80 K thermal expansion coefficients and thus thermal stresses remain low.

3.2 Discharge voltage

The apparent maximum voltage U_n for the discharge of a coil n within a system of coupled coils in an external discharge resistor R_{EL_n} is given by

$$U_{n_{max}} = I_{n_o} \cdot R_{EL_n} + \sum_{m \neq n} I_m \frac{M_{mn}}{L_m} \cdot R_{EL_m} \qquad (2)$$

with I_{n_o} the operating current of the coil n and M_{mn} the mutual inductance between coil n and m. In large coils like fusion magnets, the current I_{n_o} is quite high, $\geq 10^4$ A. In all magnets the value of the discharge resistor P_{EL_n} must be chosen high enough so that the discharge time constant $\tau_n = \frac{L_n}{R_{EL_n}}$ is of the magnitude required to limit the coil heating in accordance to eq.(1). Therefore the possible discharge voltage has a lower limit, which is rather high in large coils. Such voltages are not trivial for He-systems, because He-gas is a very poor dielectric. Therefore the voltages are especially limited to typically a few kV in boiling He as present in bath cooled magnets and at the current leads, which are usually cooled by a counterflow of He-gas.

For rapidly dischargeable magnets (energy storage coils, poloidal field windings for tokamaks), prototypes have been operated up to about 50 kV which is otherwise also the range of voltages required for regular operation of such coils. This indicates, that with careful design and pretesting, values sufficiently high for safe emergency discharges of the different kind of coils can be guaranteed.

3.3 Helium pressure increase

The superconducting windings are cooled by liquid helium (also superfluid helium) at about ambient pressure or by supercritical helium at much higher pressure, typically 5-15 bar. The whole system is therefore contained in a helium pressure vessel or, in case of forced flow cooling, the cooling ducts (sometimes integral part of the conductor) act as a pressure vessel. A sudden energy dissipation will cause helium evaporation and/or pressure increase. Therefore a venting system is required which allows helium blow down to limit the pressure increase. It must be designed in accordance with the maximum energy deposition into the helium, which can be well estimated. For this case, a helium blow off system for gas release toward the low pressure storage system will be foreseen, with a cross section chosen to keep the presssure inside the helium vessel sufficiently below the maximum design limit. As a second safety barrier, burst discs have to be installed, which have to be set at a pressure above that present at the before mentioned blow off process, but still below the maximum design limit. If the burst discs open, the helium will be usually released into the environment.

In case of forced flow cooling the helium is contained in long cooling ducts of high ratio between length L and hydraulic diameter d_h ($L/d_h=10^4-10^5$). A careful calculation of the maximum pressure increase in these ducts due to the heat load of the quench is necessary to design the proper duct thickness and helium blow off conditions. Computer programs for these calculations have been developed and are experimentally proven |7|.

The other case of possible energy deposition into the helium is that of a sudden breakdown in the thermal insulation (e.g. fast vacuum breakdown). Based on experimental results |6| heat fluxes between 0,6÷3,8 W/cm^2, dependent on the cryostat shield design (superinsulation, cooled shields, uninsulated vacuum space) can be taken as design criteria for the helium evaporation and therefore for the blow off and burst disc cross section design.

3.4 Asymmetric mechanical forces

In plasma confinement systems, the magnetic field is produced by a series of magnetically coupled coils. If one or a few coils fail or quench, abnormal or asymmetric load conditions are present. The calculations of stresses and strains in large fusion magnets are usually done very carefully with three dimensional finite element analysis. Therefore there exists sufficient confidence that also abnormal operating conditions can be taken by the structure if its loading condition is well considered in the design, based on such calculations.

3.5 Inductive current increase

In case of coupled coils, a discharge of any single coil enhances the current in the other coils due to inductive coupling. Knowing self- and coupling-inductances as well as discharge time constants (resistances), the coupling effect can be calculated by simple analysis. The current enhancement factor will often be too high to accept single discharge of the failing coil only. That means, if one coil fails it might become necessary to discharge the whole coil system. Due to the criteria discussed above, such a discharge can be conducted safely.

3.6 Experimental proof and conclusion

The above comments should have demonstrated, that the events connected with a coil quench are non-destructive through proper coil design, justifying the definition that a quench is just an abnormal operating condition which can be taken by the system without a damage.

An experimental demonstration of such design criteria on an appropriate scale concerning fusion magnets, will be provided within the "Large Coil Task" (LCT), an international superconducting tokamak magnet experiment in the frame of the IEA |8|. Its aim is to demonstrate feasibility and safe operation of large superconducting toroidal field magnets to provide a basis for the design of coils for tokamak experiments and later reactors. Full participation of industry is included to ensure that their know how for subsequent serial manufacture of such magnets is available in time. The experiment consists of a compact torus arrangement of six D-shaped toroidal field coils with 3,5 x 2,5 m inner bore each and a maximum field of 8 T. They are located in a big common vacuum tank equipped with a liquid nitrogen shield for thermal insulation. The facility is under construction at the Oak Ridge National Laboratory, three coils are developed by US-industry, the other three coils are developed by the other participants Japan, Switzerland and Euratom. For Euratom, our laboratory, KfK has taken over the responsibility and has selected the Siemens company as industrial partner. LCT will come into operation in 1982 with pretests of the first 3 coils and late 1983 as the full 6 coil compact torus.

4. *Accident situations*

Contrary to the abnormal operating conditions for which the magnet systems of a fusion device must be designed, accident situations are associated with the generation of high power arcs driven by the energy stored in the magnet system. The time scale, geometric spreading, power, and energy balance of the arcs may be used for first estimates of their consequences and damaging potential. Even in the inductively coupled magnet systems of fusion devices, the stored energy of the single magnet affected by the arc gives a realistic upper limit for the energy to be dissipated.

Three types of accidents with arcing can be defined, according to their damaging potential:

- Arcs along or across current leads or due to the rupture of a single conductor. This arc has the nominal current of the magnet and is called "Single Current Arc";
- "Multiple current arcs" are formed if more than one conductor turn ruptures, e.g. due to the heat generated in an initial single current arc. Ultimately, all ampere-turns of a magnet could be flowing in a multiple current arc;
- "Missile generation" will be the consequence of two multiple current arcs at different sections of a winding. The piece of winding between the arcs is accelerated by Lorentz-forces as long as the arcs burn. If it is unrestrained, it constitutes a damaging projectile.

4.1 Single current arcs

Single current arcs will most probably be experienced after a malfunction of leads between a magnet and its external power supply or dump resistor. They can be caused by a rupture of one lead, e.g. at a joint, or by a short circuit between one lead and ground or between the two leads powering the magnet. These leads are generally cooled counterstream by gaseous helium, the flow of which is controlled to reach room temperature upon leaving the cryostat. Thus the warm end of the leads is a particularly critical region with respect to arcing due to the very low voltage-breakdown characteristics of warm helium gas. In principle, single current arcs can also be imagined in the magnet winding itself following a conductor rupture or the breaking of a joint, but the geometry and power levels involved in large fusion magnets will most likely develop them into multiple current arcs.

Single current arcs have been observed during the malfunction of superconducting magnets. Although no extensive equipment is usually at hand to diagnose the arcs at such occasions, available evidences can be used to obtain information on the arc voltage as the most important single parameter for arriving at time scales and power levels. An experimental evidence of this kind |9| was evaluated to give the following numbers:

Stored energy $W = 800$ kJ
Operating current $I_0 = 1000$ A
Burn time of arc = 5 s

These numbers show a time-averaged arc voltage of 160 V, well above the voltage expected from standard arc theory, e.g. AYRTON's equation |10|. The following new features in magnet arcs must be evaluated as possible causes for this observation

- High values of I_a (≥ 1 kA). Instead of the inverse relationship between voltage and current, an Ohmic rule of direct proportionality must be applied. One reason is the self-pinching effect in large plasma currents.
- High pressure in the region of the discharge.
- Presence of a vapor jet blowing out of the base points of the arc.
- Presence of gradients in pressure and magnetic field which tend to increase the length of arcs even to extinction, forcing frequent new ignitions with the associated higher voltage.

In the future, special-purpose experiments would be of great value for understanding the behaviour of high-current, metal-vapor arcs in magnetic fields.

Assuming 200 V as a typical voltage in fusion magnet arcs, their development strongly depends on the operating current of, typically, 10^4A in d.c. fusion magnets such as tokamak TF-coils or mirror magnets, and 10^5A in the pulsed poloidal tokamak coils. In both types of coils, several Gigajoule of energy are stored in the magnetic field (e.g. UWMAK III, Intor).

In d.c. magnets, the arc power would amount to $10^4\text{A}\times 200\text{ V}=2\times 10^6\text{W}$. At a stored energy of 6 GJ, the arc would continue burning for 3000s. This long duration is very unlikely. Rather, it must be assumed that the magnet winding itself would be driven normal within, typically, 20 seconds by either an outside quench triggering mechanism, by the loss of vacuum due to the arc burning through the vacuum tank, or by a loss of liquid helium. Assuming a time-averaged resistivity of $2\times 10^{-9}\Omega\cdot\text{m}$ for the stabilizing and reinforcing materials which cover about 60% of the winding cross section, the Joule dissipation in the winding amounts to

$$P_W = j_{St}^2\cdot\rho_{St}\cdot V_{St} = \left(\frac{10^7\text{A}}{0.6\times 1\text{m}^2}\right)^2\cdot 2\times 10^{-9}\Omega\cdot\text{m}\cdot(0.6\times 60\text{m}^3)=2\times 10^7\text{W} \qquad (3)$$

Thus, energy is dissipated with increasing power in the winding itself at a faster rate than in the arc.

In pulsed magnets of the tokamak poloidal field coil system, the arc power is about 20 MW due to the higher operating currents. Even in some large diameter coils of the INTOR poloidal field system, the stored energy amounts to about 10 GJ of which about 50% will be thermalized in a single current arc if the same reasoning is used as above.

Single current arcs in d.c. magnets will thus lead to moderate damage, but may cause severe damage if generated in large poloidal field coils.

4.2 Multiple current arcs

If a single current arc is generated by a conductor rupture in a winding itself, its geometric path is influenced by the magnetic field distribution which drives it to the zero-B-line present in all large magnets. It acts like an arc welding beam on the adjacent conductors.

Being under considerable hoop stresses of about $2\times10^8 N/m^2$, these conductors will also rupture as they are heated and softened. The time scale involved can be estimated by taking a single conductor cross section of 10 cm^2 and assuming that it ruptures if a length of 10 cm is heated to 1000 K. With the typical enthalpy difference for metals of 2×10^9 J/m^3 between 4 K and 1000 K, the time needed to adiabatically heat the conductor to rupture is $2\cdot10^9\ J/m^3\cdot10^{-4}m^3/2\cdot10^6 W = 0,1$ s.

With the second conductor ruptured, current and power in the 200 V constant voltage arc double, leading to an avalanche of conductor ruptures with a doubling time of 0.1 seconds.

After about 1 second, the entire magnet winding ruptures and the arc carries all the magnet's ampere-turns, typically 10^7 A. In a few seconds, the resulting multi-GW arc consumes the stored energy and evaporates between 500 and 1000 kg of conductor material. Severe damage in adjacent components of the fusion device must be expected,even the triggering of further arcs in neighboring magnets.

4.3 Missile generation

As yet another type of accident in large superconducting magnets, the generation of missiles is occasionally mentioned. It assumes that a current-carrying part of the winding becomes loose and is accelerated in the magnetic field due to the Lorentz-force. A sketch for such an extremely unlikely event is shown in Fig. 3 assuming that during the burn-time of a multiple current arc within the magnet winding, a second complete rupture of the winding occurs at a distance b from the first arc. The conductor section, now mechanically unrestrained, is accelerated in the background field B of the other coils. The arcs burn into the loose section until its ends have travelled approximately b/2 away from the remainders of the magnet. At that point, the arc rearranges to burn between the fixed remaining ends of the coil. The upper limit for the kinetic energy of the loose section may be estimated by the product of Lorentz-force and distance during which it is accelerated. Assuming constant current, one gets

$$W = \frac{1}{2} m v_f^2 \le \frac{1}{2} b\cdot B\cdot I\cdot b \qquad (4)$$

with v_f the final velocity and I the total ampere-turns flowing in the section.

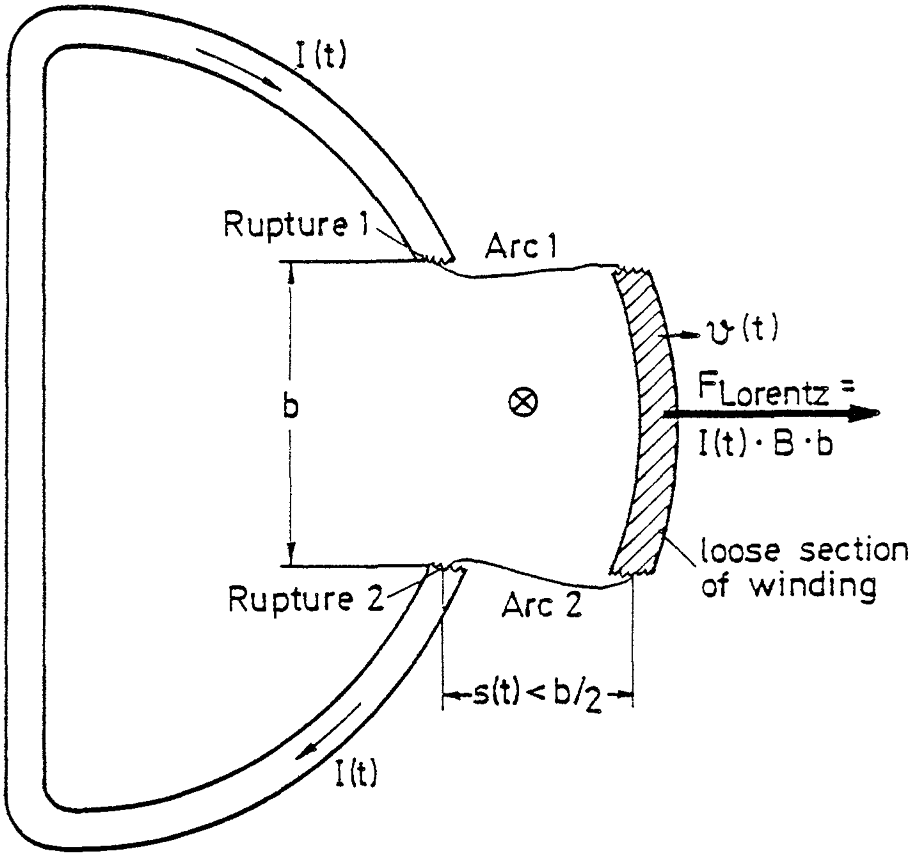

Fig. 3: Modell for double rupture of the winding with missile generation.

The scaling of W with b^2 indicates in particular that small pieces of conductor that might break off during the burning of a multiple current arc will not constitute hazards in addition to those of the arc's plasma beam. For a hypothetical missile to be of considerable impact, the extremely unlikely simultaneous winding rupture at places at least one meter apart must occur.

A missile of 2 m length would reach a final velocity of about 100 m/s after 20 ms of acceleration and contain a kinetic energy of about 200 MJ. The remaining stored energy of the magnet is thermalized in the multiple current arc replacing the missile section. With their occurrence being highly hypothetical and with their conceivable hazard potentials comparable or less than those of other accidents (multiple current arcs, airplane crashes, lithium fire), missile generation from superconducting magnets needs not to be considered usually in the safety analysis.

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DISCUSSION

J.G.CROCKER: You feel that we can design the reactors for very low probability of failure for these very large magnets as long as they are well engineered and not run in the laboratory. If one examines the data base that currently exists on the failure of superconducting magnets one finds a failure rate of about 50% per year. These are studies that were done at Brookhaven National Laboratory. A lot of small magnets that were run by physicists in the laboratory were examined. For the kinds of magnets you were talking about the data base we deal with is completely inappropriate. In the last hypothetical gross accident that you described was a slight dependance on the arc voltage. Normally one thinks an arc being generated at above the KV level. However if one initiates an arc at that very high voltage it will take much less voltage than to keep the arc driving. It takes a high voltage initially but a much less voltage to keep the arc going, or even for initiating a second strike as the path is ionized.

P.KOMAREK: First, you mentioned correctly that the magnets examined for failure probability are rather small units for physics experiments. Such magnets are not representative concerning the subject we are discussing here, because they are built mostly without safety margins (to be cheaper and to reach maximum performance data), which need not to be emphasized for the limited lifetime of such experiments. For magnets of large size built so far, like bubble chamber magnets, the same kind of stability and safety margins as discussed in my presentation have been used for the design and no serious failure has been reported yet of anyone of these magnets, partially in operation now since about one decade. For fusion magnets we use now even more accurate design tools, like finite element force calculations and should gain therefore additional confidence in reliable magnet performance.

J.SCHNEIDER: For the theoretical analysis you pointed out the fact of missing data on the voltage current characteristics of the arc, the famous Ayrton equation. Do you agree that an experiment, under realistic conditions in a fusion environment, would be necessary for determining this characteristic?

P.KOMAREK: The problem is how good one can simulate the realistic condition. One should have helium gas as an environment and one should have the right order of magnitude of current. This, I guess, makes the experiment not simple, but it should be done somewhere. Otherwise our data are very rough and we can only rely on some experimental data which were obtained by chance and which were not very carefully derived. Experiments are not built to measure carefully accidental voltages, time constants etc. and usually the data can not be quoted. There should be some safety experiments be done specific for that purpose.

J.G.CROCKER: One of the things that you pointed out there very strongly illustrates the quandary of a person who worries about fusion safety. The last accident that you postulated could be a very damaging accident to s fusion system. With such a missile you could certainly rupture most of your tritium lines. Very likely would you disrupture your cooling systems for your blanket and first wall and have a potential for activation products and tritium release. If one could postulate that this could breech the containment of the building, then one has a potential for an accident that is really beyond the design basis. Most of the people claim that this is an incredible accident. It borders on the impossible and one should not be worried about it. I like to believe that. Unfortunately we do not have a good solid data base to confirm our ability to calculate what is really going to happen.

P.KOMAREK: The first question that must be addressed is under which circumstances can such a double break of the winding occur. From the magnet point of view I usually do not see a possibility how this could occur within that short time of a couple of seconds. There must be a coupling with the transient external failure. Then it is really a severe damage of the whole reactor system which occurs simultaneously with the magnet damage. The last safety barrier which you can use is the containment. I would claim that in any case the containment should and could be designed in such a manner that this impact can be taken without a break in the containment.

W.DÄNNER: It was new to me that the reason for a missile production has to be a two-fold arc. I am not an expert in this field but I could imagine that also, as a consequence of a single arc, a deformation of the coil can occur. May be also through a similar process like a missile production. This deformation might be such large that also external components of the reactor could be destroyed.

J.G.CROCKER: This could lead to severe imbalance in the forces.

P.KOMAREK: You mean that the coil will be overloaded by the asymetric forces and perhaps some neutral injectors between two coils could be quenched?

W.DANNER: And the coil would break.

P.KOMAREK: It would break only on one cross section without missile generation.

W.DANNER: But with deformation.

P.KOMAREK: You get a deformation.

W.DANNER: And this deformation might cause accidents in cooling systems, tritium systems.

P.KOMAREK: I did not exclude that. I said that deformation and arc burning would cause damage to other reactor components but not missile generation.

J.GRUBER: When such a coil breaks at its cross section, does it expand quickly in just a second or so?

P.KOMAREK: No, I would say it is over one second because the structure is very thick, 20 or 30 cm of coil case outside. If one has this burning cross section, within about 1 or 2 seconds in the pessimistic case, then it is a question of where it occurred, how much magnetic energy is still available to overbend the system and how fast the temperature increases in the other parts of the system so that the mechanical strength is decreased. But roughly within 5 seconds this coil will dump its whole energy. The other coils will not be ruptured and usually go down slower on their own discharge within about 30 seconds or even more. But as soon as this one coil has no current, no mechanical force, it is just a question of these few seconds that overbending can occur. Hypothetical missile production would mean that a whole piece must turn so fast that it breaks the outer part immediately. It might be an interesting exercise to look, with a computer code, at how the thermal energy can distribute in such a piece and temperature increase occur, and how fast it softens. Could this occur much faster than the energy dump or not. I would be careful in saying it is not possible; I have not considered it quantitatively.

J.G.CROCKER: You just hit a point which has somewhat disturbed me. In the USA INTOR design we originally had coil cases with about a thickness of 20 cm. But the latest design comes out with much thinner coil cases of about 5 cm thickness. Originally I have been doing safety thinking on the basis of very thick coil cases that were going to be essential missile barriers and were going to contain things in them. Suddenly our designers came up with 5 cm thick cases, which poses an entirely different situation from the point of view of structural containment ability.

P.KOMAREK: I have not seen that design yet so I cannot discuss it. However you can add structural supports outside.

J.CROCKER: Yes, the INTOR team came up with very elaborate external support for that particular coil design.

W.DANNER: What is a typical temperature of an arc and what does it depend upon?

P.KOMAREK: It depends upon the gas which is usually copper or steel vapor. It usually depends not so much on the arc power, it usually depends more or less only on the arc material, on the environment, and the usual order of temperature of such an arc is 5000-6000° C.

M.S.KAZIMI: With respect to the wall moderator type of event which we have also considered: Basically your recommendation is to design the coils in such a way that they can take the quench both through the external resistance or through thermalization and hence one has double protection, which I think is reasonable. However, you have not discussed some other patterns of connections which perhaps would distribute the forces somewhat more easily in the system if you were to discharge every other coil for instance instead of the total number of coils all together or the single faulted coil.

P.KOMAREK: If one monitors a quench in one of the coils then, in a Tokamak for instance, one would discharge the whole system. I did not say that the whole system is discharged into one unit. It would depend on the kind of circuits one would have. For instance in INTOR they are discussing burn control by ripple changes - ripple changes by decreasing the currents in TF coils, each second one or so for instance. One must have in this example at least two separate charging and discharging circuits for the torus. One can think about an even more sophisticated distribution of electric circuits over the torus, and then one has more circuits to treat. In all these cases then one has to look at the inductive balance and also at the force balance.

J.G.CROCKER: If one had a combination of a series and parallel connections, for example perhaps every other magnet connected in series. one might have two separate series which would balance the forces at all times in that case.

M.S.KAZIMI: With respect to the more serious event which I also happen to be interested in pursuing: Does one have to consider all the energy sinks available at that particular moment? Have you done a calculation as to how much of the coil energy would remain to be exhausted in the missile generation?

P.KOMAREK: We did it very pessimisticly. We did not take into account additional energy sinks. This model for missile generation takes into account that the coil is simultaneously discharged through the arc, internally discharged, and one has a final velocity after about 1 m distance. At that point the arc is shifted between the remaining ends of the coil and not through the loose section. The piece flies away but is not accelerated further.

M.S.KAZIMI: That is true but the total energy is now being given to that piece of coil that is being accelerated?

P.KOMAREK: The coil energy will be completely dumped in the remaining internal arc, otherwise the energy balance would not be fulfilled.

M.S.KAZIMI: There are natural things that will decrease the available energy. The breakage itself will consume substantial amounts of energy.

P.KOMAREK: The breakage is due to the coil energy itself. It could however be caused by an external event. This is a more pessimistic case. Usually one has a more relaxed case.

M.S.KAZIMI: Yes, I realize that this is one of the more severe cases and that is why I am interested in pursuing it. I believe it is good to think of the maximum possible but also of the more mechanistic case of doing things simultaneously.

T.K.ALLEN: It seems that the arc is self extinguishing and it is hard to believe that the arc could be of that length within an enormous magnetic field. It appears like an ideal self extinguishing system and I would expect that to happen rather quickly, as soon as a slight deformation has occured.

P.KOMAREK: Arcs may be selfextinguishing or blown out magnetically if they are driven on a given maximal voltage. In case of a magnet the stored energy will adjust the voltage to keep the current flowing until the energy is dumped. That was the reason for saying that a good numerical equation about the arc should be determined experimentally.

T.K.ALLEN: The practical experience of people who extinguish arcs would be of particular benefit.

P.KOMAREK: However these people always tell that one can not extrapolate their findings to this completely different material (helium, magnetic field etc). They always refuse to give data which one could use in general.

J.G.CROCKER: Some large experiments might be helpful in the future. Larger coils are being built for the large coil programme. After these have gone through the set of planned experiments, near the end of that programme, I would like to see some experiments along the line that could potentially be very damaging to the coils. Could we propose reasonable experiments that could be carried out while properly instrumented and thus get valid data? I would like to see something like that and would like to have your reaction on what sort of experiments should be done.

P.KOMAREK: You are not the first one suggesting that. We are discussing this possibility in the Executive Committee for LCP. However, the coils are very expensive and one would like to use them later on in other experiments as well. One could say let us at least destroy one of these coils, but the question then arises which coil? This discussion is not yet finished.

J.G.CROCKER: But you as a magnet designer and builder would support this kind of research?

P.KOMAREK: We are thinking seriously about that and it is not decided yet whether we will do this as a final experiment in LCP or not. It might be done. The input from this workshop might be useful to look at that.

J.G.CROCKER: We would be quite happy to lend you support on it because small scale experiments such as were done in the past, never completely answer the questions for the large scale conditions.

J.GRUBER: What happens to a helium system in a single current arc, in a quench?

P.KOMAREK: It evaporates very quickly and without further damage to other systems, taking evaporation pressure into account and escapes through the vent line.

SOME ENVIRONMENTAL ASPECTS OF FUSION REACTOR WASTE: POTENTIAL ENVIRONMENTAL CONCENTRATIONS OF RELEVANT RADIONUCLIDES IN DRINKING WATER

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Abstract

If traces of 1 ppm of any element are irradiated in a flux of 14 MeV neutrons corresponding to 1 MW/m² (4.5×10^{13} $cm^{-2}s^{-1}$) or by an equal flux of thermal neutrons, an important contribution to the long lived ($T > 10^2$ a) overall biological hazard potential of all impurities stems from the activated metals of the following three groups:
(a) Mn, Fe, Ni, Pb, Bi; (b) Be, Al, Zr; (c) Mo, Tc.

An evaluation of the potential hazards originating from those radionuclides for man should not only consider the activities and their radiotoxicities alone, as does the biological hazard potential, but should also include properties that describe the mobility of the radionuclides in the environment. Here it is proposed to use the solubilities of compounds of the radionuclides in pure water as a first approximation.

For relevant environmental E_H/pH domains (E_H = electrochemical potential) group (a) metals are potentially soluble releasing positively charged ions (cations) in aqueous solution. Group (b) metals have a low solubility, but form again cations. Group (c) metals are readily soluble forming negatively charged complexes $(H)MeO_4^-$.

Using the specific activity generated in the neutron fields mentioned above one can convert the solubility into a potential contamination level. Based on the recommendations of the International Commission on Radiological Protection (reference) levels of contamination are defined, which may be accepted for drinking water. The potential levels for Mn, Be and Zr lie below the reference levels, whereas Ni, Mo, Bi, Al and Pb lead potentially to an excessive equilibrium contamination of pure water.

Measurements of the concentrations of metal cations in equilibrium soil solutions have yielded values that are lower by many orders of magnitude than the ones in pure water. If calculations of potential radioactive contaminations take into account the weathering conditions (i.e. the additional ions present in soils) and the interaction of the contaminant with soil, that discrepancy may be removed. A model originally formulated for the description of the interaction of metals with sediments in the aquatic environment is proposed for describing the given situation.

(I) INTRODUCTION

The high neutron flux in a fusion reactor will activate the materials. A fraction of the radionuclides mobilized e.g. during maintenance operations or from a repository not located in a deep geological formation will enter the biosphere. A conventional way of evaluating the activities with respect to their hazard for man is biassing each activity with its respective radiotoxicity as given by the maximum permissible concentration (MPC) of the soluble species of the radionuclide in water as recommended by the International Commission on Radiological Protection (ICRP, 1959). Water is chosen as carrier, whenever it is assumed that the main path of the radionuclide to man is either drinking water or food. It is assumed that the concentration in all types of food is the same as in the water of the environment, in which the food is grown. Furthermore the passage of the radionuclide through the walls of the small intestine into the body fluids is assumed not to depend on whether it has entered being dissolved in water or as part of either plant or animal tissue. Since the pathway to man is not included in the biassing factor MPC for the activity, the hazard calculated in this way is a potential biological hazard (BHP) inherent in the radionuclide inventory.

$$BHP_i(t) = \frac{\text{activity}_i(t)}{MPC_i} \qquad (1)$$

i = radionuclide
t = time after end of irradiation

The time dependent function $BHP_i(t)$ can be converted into an index by integration

$$PHI_i = \text{constant} \times \int_0^\infty BHP_i(t)\, dt \qquad (2)$$

The potential hazard index (PHI) is a measure of the potential hazard accumulated during the entire radioactive live of the inventory.

In the past those two quantities have been employed frequently to compare the radioactive inventories of fission and possible fusion reactors (e.g. Häfele, 1976).

In this paper a ranking of the potential radionuclides generated in the first wall of a fusion reactor is performed using their biological hazard potential.
Measuring is abandoned and the potential equilibrium concentration of dominant radionuclides in environmental water is calculated based on models and data commonly used in environmental sciences. The potential concentrations will be compared with the permissible concentrations as calculated using the international recommended data (IRCP, 1959, 1979, 1980).

(II) SELECTION OF RADIONUCLIDES

In order to eliminate the specific features of the fusion reactor designs,only the minimum activity is calculated,which is common to all designs. That is the activity generated in the first wall by activation reactions,which are based on the primary fusion neutrons, i.e. the 14 MeV neutrons. For low wall loads ($\leq$ 5 MW/m²) the generation of long lived ($T \geq 10$ a) activity is nearly independent of the neutron flux ($\emptyset$), i.e. it is a good approximation to assume that the radionuclide inventories are determined by the neutron dose ($\emptyset t$) only.
We normalize therefore the neutron flux to 1 MW/m² (corresponding to 4.5×10^{13} $cm^{-2}s^{-1}$). Since breeding of tritium is essential at least for the first generation of fusion power reactors, and since some designs consider breeding reactions involving mainly (epi-)thermal neutrons, also the occurrence of thermal (non-threshold) activation reactions has been considered, assuming again a thermal neutron flux of 4.5×10^{13} $cm^{-2}s^{-1}$.

It was assumed that 1 ppm of any element of the periodic table be present in the material for the first wall and thus exposed either to fast (14 MeV) neutrons or thermal neutrons.

Details and the results of the calculations can be taken from Tab. 1 and its associated comment. It is emphazised that only first generation radionuclides have been considered, to be consistent with the concept of calculating the lower boundary of the activity in the fusion reactor structure.

The activation reactions taken into account were the ones, for which cross sections are available in the compilation of Alley and Lessler (Alley, 1973): Those are neutron capture (n,γ), (n,2n), (n,α), (n,p), (n,pn), (n,H^3).

In Tab. 1 reactions leading to a biological hazard potential lower than 10^{-11} m^3 of water per cubiccentimeter of first wall material have been eliminated.

To give a visual impression of the time period, during which the hazard prevails, Fig.1 has been constructed. Noble gases are left out. The chemical symbol of the mother and - if different - the daughter i (given behind a slash (/)) is placed in the position (y,x) = (BHP_i, T_i) within the coordinate system. The elements involved in fast activation reactions are underlined in contradistinction to the ones involved in thermal reactions. Since some elements have radionuclides with several half-lives, the chemical symbols of those appear several times within the plot.

As one can see, each radionuclide plays a role during a time of less than ten half-lives, after which the potential hazard of a longer lived nuclide becomes more prominent.

The results can easily be modified to apply to material compositions different from those chosen, since the calculated BHP_i is proportional to the concentration of the mother nuclide So one can see that in 1 cm^3 of a first wall consisting mainly of Mo the amount of long lived Tc99 produced by the Mo98(nγ)Tc99 reaction will be about an order of magnitude less than the amount of Tc99 produced within 1 cm^3 of a light water reactor (LWR) fuel element by the fissioning of uranium. Note that the potential biological hazards due to the actinides generated in LWR-fuel shown in Fig. 1 were divided by 10^4.

There is another application possible for the data displayed in Fig.1, again allowing a comparison between the activity generated in a fusion reactor and in a LWR: In the INTOR-blanket design (Abdou, 1981) a Pb-layer of about 5 cm thickness is used for neutron multiplication. A wall load of 1 MW/m^2 corresponds roughly to a thermal power output of 22.4/14.1 MWt/m^2. The reciprocal value of that is roughly 6 cm^2/kWt. The amount of Pb corresponding to 1 kWt is therefore 6 x 5 cm^3 = 30 cm^3. If one neglects the decrease of the fast neutron flux within the first wall and within the multiplier (a rather crude assumption, but the order of magnitude will still be right), one can calculate the potential biological hazard due to the very long lived Pb activation

(2×10^{-1} m³/kWt) and relate it to the potential hazard due to Np in a LWR-fuel element that has delivered the same amount of energy, i.e. 2 kWta. According to Fig.1 the long lived potential hazard due to Np generated in a volume of 1 cm³ of LWR-fuel (that corresponds roughly to an energy of 2 kWta (basis: 3×10^4 MWtd is generated per ton of uranium, i.e. per 5×10^5 cm³ of uranium)) is 2×10^3 m³.

The location of the relevant elements (mothers and daughters) within the periodic table is shown in Tab. 2: They are printed there using bigger letters. Additionally the elements that are situated in the upper half of Fig.1 are put into squares. Some radionuclides will lead to relevant potential hazards only, if the corresponding mothers are employed in larger amounts, such as the neutron multipliers Pb or Be, or the mother/daughter doublets Fe/Mn or Mo/Tc. They are encircled, to emphasize their potential importance.

The high hazards associated with the elements marked by the superscript° might be due to the assumed high radiotoxicity (MPC = 3×10^{-5} Ci/m³). It had to be assigned to them according to the recommendations of the ICRP: For those radionuclides neither the old (ICRP, 1959) nor the improved metabolic models (ICRP, 1979, 1980) have yet been employed to calculate the maximum permissible concentrations in water or equally useful values, i.e. the annual limits for intake (ALI).

As one can see, the environmental and metabolic behavior of the relevant activation products (except Tc) can be described using the knowledge already gathered on their stable isotopes. Contrary to this the amount of data for the description of the actinides, the long-lived waste nuclides of fission reactors, is smaller.

Some general properties of the relevant elements, i.e. their solubility and toxicity, are already specified in the table, to give a rough guideline to the relative importance of the selected elements (symbols #, i, r, a and underlining): Whereas the concentration of chemically toxic or very insoluble elements in water will always be limited, other radionuclides, the stable isotopes of which are essential elements, very rare or non-toxic, might be present in water in high concentrations. That should be taken into consideration in selecting potentially hazardous elements.

In the following paragraphs the heavy metals (atomic weight higher than 40) among the relevant elements will be dealt with in more detail, to assess their concentration in drinking water. Their concentration depends on the chemical species, in which it appears in environmental water. Which species will actually be present in equilibrium, will be determined by

(*) the water chemistry (see Chapter III)

(*) the metal transfer between the aqueous and solid phase in soils (see Chapter IV)

(III) WATER CHEMISTRY

Aquatic chemistry is concerned with the chemical behavior of natural aquatic systems of all kinds, in order to define the pertinent variables governing the composition of natural waters.

In the following it will be determined using equilibrium thermodynamics, which chemical species are energetically possible. In Chapter III,1 the principles are briefly outlined: The chemical potential and a thermodynamical measure of the concentration, the activity, is described. Then the electrochemical potential E_H is explained, which is a measure of the reducing or oxidizing power of an environment. Readers already familiar with those subjects might skip these chapters, therefore.
In Chapter III,2 electrochemical equilibrium diagrams (E_H-pH - diagrams) will be used to indicate as a function of the solution pH and E_H the thermodynamic limits of stability of the metal in relation to its ions, to the ions of the water and to the reaction products of these ions (hydroxides, oxides, complexes).

(1) Principles: (a) chemical potentials

A basic quantity in the thermodynamics of e.g. water systems is the chemical potential μ_i of a chemical species M_i. It is used analogously as is the potential energy in physics: A system develops in the direction, in which the sum of the energies of the components decreases. The potential μ_i is split into two parts:

$$\mu_i = \mu_i^0 + RT\ \ln(M_i) \qquad (3)$$

(M_i) is called the chemical activity, in case M_i is a solid or liquid, and it is called fugacity, when M_i is a gas.

The magnitudes of μ_i^0 and (M_i) are arbitrary in the sense that only their sum μ_i is fixed (Stumm and Morgan, 1970,p.29).

There is a connection between the dimensionless quantity activity (M_i) and the concentration C_i(mole/l): In infinite dilute solutions both quantities are numerically equal by definition. The degree of dilution within an aqueous solution is characterized by the ionic strength I

$$I = \frac{1}{2} \sum_l C_l z_l^2 \qquad (4)$$

l = all ionic species in solution

where C_l is the concentration of the ionic species l,and z_l is its electric charge (dimensionless number). For $I \leq 10^{-2}$ mole/l (typical for water in streams, lakes) the deviation between the numerical values of activity and concentration is usually less than a factor of 2 (Stumm and Morgan, 1970, p. 80 ff.).

μ_i^0 and (M_i) are defined consistently with respect to one another, according to the following conventions that define the standard state:

$$\text{standard state: (superscript }^0) \quad \left\{ \begin{array}{l} (M_i) = 1 \\ \\ \mu_i = \mu_i^0 \end{array} \right\} \begin{array}{l} \text{for the pure state of a solid} \\ \text{or liquid at 1 atm pressure and} \\ \text{for an ideal gas (practically} \\ \text{all common gases) or an ideal} \\ \text{solution at 1 atm (relation-} \\ \text{ship between ideal and real} \\ \text{solutions is described in} \\ \text{Garrels and Christ (1965))} \end{array} \qquad (5)$$

The standard chemical potentials of the elements (in their stable configurations) are taken to be 0 by convention, because chemistry does not deal with the decrease or increase of the intrinsic energy of a system, when one element is transmuted into another (e.g. Hg into Au).

$$\mu_i^0 = 0 \quad \text{for the standard state of the elements} \qquad (6)$$

Based on the concept of chemical potential the compositional changes of a chemical system are described by the following basic laws of thermodynamics.

Let us consider the reaction

$$aA + bB + \ldots = cC + dD + \ldots \qquad (7)$$

abbreviated as

$$\sum_i \nu_i M_i = 0 \qquad (8)$$

ν_i = a,b,c,d... are the stoichiometric coefficients (positive are the ones (c,d...) on the right, i.e. the ones for the products, negative are the ones on the left, for the reactants). M_i = A,B,C,D are the species or phases involved, they are usually compounds. The chemical potential of a compound, sometimes called (Gibbs) free energy, is calculated from the energy change accompanying its formation from the elements (see below).

The chemical equilibrium is arrived at (i.e. no further chemical reactions will take place), when no further decrease ΔG of the chemical potential of the system can be achieved:

$$\Delta G = \sum_i \nu_i \mu_i = 0 \qquad (9)$$

$$= \sum_i \nu_i \mu_i^0 + RT \sum_i \nu_i \ln(M_i) \qquad (10)$$

$$= \sum_i \nu_i \mu_i^0 + RT \ln \frac{(C)^c \ (D)^d \ldots}{(A)^a \ (B)^b \ldots} \qquad (11)$$

i = products and reactants

If the μ_i^0 are available from the literature, one can calculate the ratio K of the activities (called "equilibrium" or "stability constant")

$$RT \ln K = -\sum_i \nu_i \mu_i^0 = RT \sum_i \nu_i \ln(M_i) \qquad (12)$$

Widely used compilations of the standard free energies μ_i^0 are the ones of Latimer (1952) and Rossini et al. (1952). Sillén and Martell (1964) have compiled stability constants of metal-ion complexes.

Principles: (b) pH and E_H

The aqueous environment is frequently described by specifying

(a) how acid or alkaline it is, i.e. the activity of the protons is given, regardless where the protons stem from (pH = -log (H^+))

(b) how reducing or oxidizing it is, regardless, which species are the oxidizing or reducing agents within the water. Reduction is basically the addition of one or more electrons. So the ability of an aqueous solution, to oxidize or reduce e.g. the heavy metals is measured by giving the activity of the electrons ($p_\varepsilon = -\log (e)$) (Stumm and Morgan, 1970).

The activity of the electrons is - as any chemical activity - connected with their chemical potential. It is taken to be 0 as will now be explained:

Actually electrons do not appear explicitly in a reaction, where the species dissolved in water oxidize or reduce e.g. heavy metals, so their chemical potential is not needed in thermodynamic calculations. An example is the important reduction reaction that converts insoluble Fe (Fe^{3+}) into the soluble form (Fe^{++}).

$$Fe^{3+}_{aq} + \frac{1}{2} H_{2g} = Fe^{++}_{aq} + H^{+}_{aq} \qquad (13)$$

(subscripts designate the phase of the species: aq = dissolved in water, g = gaseous, s = solid).

The electron of hydrogen is caught by iron:

$$Fe^{3+}_{aq} + e = Fe^{++}_{aq} \qquad (14)$$

The reaction delivering the electron is

$$\frac{1}{2} H_{2g} = H^{+}_{aq} + e \qquad (15)$$

The division of reaction (13) into the "half-reactions" (14) and (15) is common practice, but still artificial (see Pourbaix, 1966, p. 57). The convention associated with that artificial procedure is the above mentioned removal of the artificially introduced degree of freedom for the chemical potential of the electron.

$$\mu^0_e = 0$$

Consequently also the standard chemical potential $\mu^0_{H^+}$ of the proton is zero. The change of the chemical potential of the system accompanied by the reaction (13) can therefore be calculated from the change of the potential due to the half-reaction (14) alone.

In general reactions involving reductions or oxidations (redox reactions) can be written

$$\sum_i \nu_i M_i + ne = 0 \qquad (16)$$

(sign conventions as in (8))

The change of the potential is given by (10) and (11).

The change of the chemical potential accompanying the transfer of electrons is identical with the work done in transporting electrons against the action of an electric field E_H, which can be explained by the following example:
Consider the deposition of dissolved copper

$$Cu_{aq}^{++} + 2e = Cu_s \qquad (17)$$

Let us assume for simplicity that the reaction proceeds in the following experimental setup: A piece of pure copper dips into a solution of cupric ions (Cu^{++}), into which also an electrode of (inert) platinum dips, surrounded by bubbling hydrogen gas at 1 atm (the "reference electrode"). The system proceeds to a state of lower potential energy, while electrons are pulled out of the piece of solid copper. The energy gain ΔG is converted into the work necessary to move the electrons from the reference electrode against the electric potential E_H, measurable with a voltmeter between the copper and the reference electrode:

$$\Delta G = nFE_H = \sum_i \nu_i \mu_i^0 - RT \ln \sum_i \nu_i (M_i) \qquad (18)$$

$$= nFE_H^0 - RT \ln \sum_i \nu_i (M_i) \qquad (19)$$

n = 2 is the number of electrons transferred from the reference electrode to the copper, a constant F with numerical value $\neq 1$ is necessary, when ΔG is not measured in units of eV, but in calories: The numerical value of F equals the amount of calories (i.e. 23060) necessary, to pull 1 mole of electrons (i.e. a charge of 1 Coulomb) against the action of an electric field of 1 Volt.
In analogy with the word for ΔG, E_H is called "electrochemical potential".

It should be noted that the definition of the sign of E_H depends on the sign convention for the stoichiometric coefficients ν_i. Here the coefficients of the reduced species are taken positive and the ones of the oxidized negative, following the sign convention of Stumm and Morgan (1970), Garrels and Christ (1965) or Pourbaix (1966), the so-called European convention:

$$\text{oxidized state} + ne = \text{reduced state} \qquad (20)$$

$$E_H = E_H^0 - \frac{RT}{nF} \ln \frac{(\text{reduced state})}{(\text{oxidized state})} \qquad (21)$$

It remains to relate the activity of the electrons expressed as $p\varepsilon = -\log(e)$ with the electrochemical potential E_H

$$p\varepsilon = \frac{F}{2.3\ RT}\ E_H = E_H/0.059 \quad \text{at } 25°C \qquad (22)$$

(2) Applications: Stability of chemical species in aqueous solutions

(a) E_H - pH - diagram for water

Water saturated with oxygen at 1 atm and 25°C (the most strongly oxidizing water) is characterized by a functional relationship between E_H and pH, which can be calculated using (18) - (22) and the conventions (5) and (6) for the standard state.

$$O_{2g} + 4H^+_{aq} + 4e = 2\ H_2O_l \qquad (20a)$$

$$E_H = 1.23 - 0.059\ pH \qquad (21a)$$

Similarly, for water at 25°C, through which hydrogen (at 1 atm) bubbles (i.e. saturated with hydrogen or the most strongly reducing water) one has

$$2H^+_{aq} + 2e = H_{2g} \qquad (20b)$$

$$E_H = -0.059\ pH \qquad (21b)$$

(21a) and (21b) define the range of possible aquatic environments ("water stability limits"). Baas Becking, Kaplan and Moore (1960) have accumulated essentially all the measurements of (E_H/pH) available in the literature. Fig.2, from their work, shows these measurements and serves admirably to illustrate both range and frequency of the environments. Fig.3 (Garrels and Christ, 1965) shows the regions of greatest frequency of readings for several types of natural waters.

> "The general picture that emerges is that only surface waters with good circulation are oxidizing, whereas confined waters rapidly lose their oxygen content, whether confinement results from fixation in rock or soil pores, or by prevention of overturn of open waters. Organic-free waters lose their oxidizing character by reaction with silicates containing ferrous iron, such as biotite, chlorite, amphiboles, pyroxenes, or by contact with sulfides or ferrous iron-containing carbonates. There are many other possible inorganic reducing agents, but the iron compounds are probably

the most important quantitatively. The pH tends to rise because of hydrolysis of the silicates, so that the environment becomes alkaline as well as reducing. In environments containing organic matter, biochemical reactions quickly remove oxygen, commonly with marked increase in CO_2, and with production of hydrogen sulfide. The influence of bacteria is paramount (the reader is referred to Bass Becking, Kaplan, and Moore (1960), for detailed authoritative discussion of the role of microorganisms in controlling environment) and deoxygenation tends to be accompanied by pH lowering as CO_2 and H_2S are generated. Some bacteria even release hydrogen, and potentials may sink close to the water stability limit. Below the water table the environment can be assumed to be alkaline and reducing, except in local instances of high rates of water flow. The best evidence is the persistence of sulfides in ores for millions of years, without destruction of the finest details of surface characteristics." (taken from Garrels and Christ, 1965)

(b) Stability of metal radionuclides

The environmental behavior of the long lived radionuclides generated by activation of the potential structural alloy constituents or neutron multiplier materials (encircled in Tab.2) will be described within the framework of equilibrium thermodynamics explained above.

A first step in this direction is the investigation of the systems Me-H_2O at 25°C and 1 atm total pressure (Me = encircled metals in Tab. 2).

The basic formal procedures employed are - for more details see Pourbaix (1966), p. 46 -

(1) establishing of a list of known species (A,B,...) that exist in the system Me-H_2O and finding the values of their chemical potentials μ_i^0 (i = A,B,...)

(2) writing of the chemical and electrochemical reaction equilibria, in which these substances (A,B) can take part two by two (see (8) and (16))

$$aA + cH_2O + ne = bB + mH^+ \qquad (23)$$

(3) using of (12), (18) and (19) to determine rations (A)/(B) of activities. With the conventions (5) and (6) from the ratios

- (*) either the E_H-pH dependence of the activity of the dissolved substance B in equilibrium with the solid A can be calculated (to yield lines of equal activity in the E_H-pH - diagram)
- (*) or the border lines in the E_H-pH - diagram can be calculated, for which (A) = (B). These lines enclose the E_H-pH regions of relative predominance of one dissolved substance over all the others.

The compilation of the corresponding E_H-pH - diagrams (Fig.4a-l) is taken from the Atlas of Electrochemical Equilibria in Aqueous Solutions (Pourbaix, 1966).

The domain, where water is stable, lies between the broken lines a and b. The rest of the broken lines are the borders of the regions, where the dissolved species - printed in that region - predominates over all the other considered chemical species in solution. (The numbers refer to the equations of the type (23), from which they were constructed.) The solid substances considered are printed in heavy type, and their fields of stability are bordered by heavy lines. They dissolve in pure water producing dissolved species, the activity of which depends on the electrochemical potential E_H and pH of the aqueous system. The lines of normal printing in the diagrams are the connection of the points of equal activity in the aqueous solution.

Every line carries a number (in circle) that refers to the corresponding number of the equation of type (23), from which the line was calculated. The lines of equal activity were plotted for the activities 10^{-6} moles/l, 10^{-4} moles/l ... 1 mole/l, only the exponents being written in the diagram.

In the next step of approximation the simultaneous presence of major environmental species has to be considered, e.g. CO_2 and SO_4^{--}. This means that the number of reaction equilibria has to be enlarged (see e.g. Hem, 1965, or Garrels and Christ, 1965). Fig. 5 (from Hem, 1965) is the result for the system $Fe-S-CO_2-H_2O$. New solids appear in places, where they are less soluble and more stable than the oxides and hydroxides. The area of the regions of high solubility decreases. But also the opposite can happen, if the new substances lead to the formation of soluble complexes.

So the choice of additional chemical substances predetermines the result in that respect. Therefore, the tedious calculations of the type (1) - (3) for establishing the diagrams have been programmed for computers, to enable taking into account a great many of new substances.

An example is the system computed by Morel and Morgan (1972), which comprises 20 metals, 31 ligands, more than 700 complexes and 83 possible solids and CO_2. The ligands are inorganic as well as organic ones, i.e. (CO_3^{--}, SO_4^{--} as well as EDTA). When considering only the inorganic ligands (Morel et al., 1973), it is found that the predominant species are the ones given in Tab. 3.

Since Tab. 3 does not contain soluble complexes not considered in the compilation of Pourbaix (1966), the results of those calculations (System Me-H_2O) seem to provide a guess of the upper limit of solubilities, whenever organic complexing agents can be neglected.

Here only two informations will be taken from the calculations of the system Me-H_2O,to give an example of the possible procedures in this field.

(1) Within the pH - limits of drinking water (pH5...9) the solubility of the metals (except Be, Al, Zr and most probably Nb) can exceed 0.1 moles/l.

(2) Except for Mo, Tc and to some extent Al the possible soluble ions or complexes are positively charged in neutral of acid solution.

(3) For the following evaluation of the potential impact of radioactivated metals on man via the drinking water pathway it is assumed that such water will not be drunk , if its ionic strength I due to dissolved metals is bigger than the ionic strength of sea water, i.e. bigger than 1 mole/l. One seems to be justified to assume a maximum concentration of the potentially soluble metals of C = 1 mole/l. The elements carrying the index a in Tab.2 will not be allowed to have such a high concentration. Pb, e.g., is limited by Public Health Service Drinking Water Standards (1962) to appear at a maximum concentration of 50 µg/l (i.e. 3×10^{-7} moles/l of Pb^{++}).

(3) Application: Replacement of the biological hazard potential

It appears only obvious, to use the potential environmental concentrations C obtained in water chemistry for the evaluation of the potential impact of the radioactive heavy metals on man. The bridge is the specific activity S_v of the heavy metals (Curies per gram of metal). The potential contamination of drinking water is defined as

$$C^* = C \frac{M}{\rho} S_v \qquad (24)$$

(M = molar mass, ρ = density of metal, C^* in Ci/l)

In its former philosophy the International Commission on Radiological Protection used the maximum permissible concentration (MPC), mentioned above, a radionuclide may be permitted to have in water. That concept was abandoned, one of the reasons being that environmental reconcentration or depletion may occur. But it is still applicable to limit concentrations in drinking water.

Therefore the contamination C* is biassed by those recommended maximum concentrations C_w^* in drinking water

$$\frac{C^*}{C_w^*} = \frac{CS_v}{C_w^*} \frac{M}{\rho} \qquad (25)$$

The ratio (25) is calculated as explained in Tab.4 and plotted as function of its half-life T in Fig.6.

$$\frac{C^*(T)}{C_w^*} = \frac{CS_v(T)}{C_w^*} \frac{M}{\rho} \qquad (26)$$

Note that the upper limit of solubility was taken as high as 1 mole/l.

The longer the half-lives the more the ratios (26) approach unity, the activation of Fe, Be, Zr and Nb turning out to be not even potentially radiotoxic. A second favourable fact comes out: The distance to the long lived actinides, generated in LWR-fuel elements, Am, Pu, Np, has become much bigger than, when merely the biological hazard potentials are compared.

IV METAL TRANSFER BETWEEN AQUEOUS AND SOLID PHASE OF SOILS

A field comparably wide as the one of aquatic chemistry is the action of the soil on the pore water composition. Using the very detailed review of Förstner and Wittmann (1979) in this paper only some of the basic principles will be introduced that will have to be considered in connection with radioactive heavy metals.

In Tab. 5 the carrier substances (left) and the mechanisms, which immobilize the heavy metals,(right), are summarized. Item (1) of that table has been dealt with in Chapter III, it is the field of aquatic chemistry. Items (2) - (5) describe mechanisms of immobilization. They characterize the well known and ecologically important property of soils,to purify groundwater.

The microscopic picture (Jenne, 1978) is shown in Fig. 7: The clay particles of the soil are covered by precipitates of

(*) iron and manganese, i.e. oxides, sulfide
(*) calcium, i.e. carbonate, phosphate
(*) alumosilicate

and organics, i.e. living organisms or organic carbon, all of which act as sinks for heavy metals: During the precipitation process the heavy metals are coprecipitated, even if their solubility products do not yet demand precipitation. The soil particles act mainly as mechanical substrates.

The biological availability of the transition metals in pore water is dominated by the iron/manganese-oxide sinks. They are only scarcely element specific, and the amount of substitutions in the Mn-precipitates (little 50 µ - nodules) is so large - they are microscopic "garbage bins" - that no crystals can develop. Therefore redissolution of the precipitates is easy. Contrary to them the sulfide, carbonate and phosphate are more element specific sinks. The more important sinks generally are comprised of thermodynamically metastable phases. Recrystallization of amorphous hydrous iron-oxides to goethit and hematite is particularly inhibited by coprecipitation of natural organic substances, by trace metals, by anions other than oxygen or OH and by sorption of silicate and phosphate.

Fig.8 is the experimental counterpart of Fig. 7. It shows the relative distribution of the metals over the various sinks. Tab.6 gives their absolute distribution.

A quantitative picture of the reduction of the heavy metal content in pore water is given e.g. by Faßbender and Seekamp (1976). They have shown that the Cu, Co, Ni and Pb concentrations in equilibrium soil solutions of low pH (pH3...5) forest soils are many orders of magnitude below the ones of free hydroxides, carbonates and phosphates in pure water.

V CONCLUSIONS

Potential concentrations of radioactive heavy metals in groundwater can be modelled on the basis of aquatic chemistry and the present knowledge of the interaction between solid and aqueous phase of soils. A first step in that direction would be the description of equilibrium states.
In the relatively simple systems Me-H_2O the calculated concentrations seem to be higher than in systems containing more elements as ligands. Furthermore because of their chemical toxicity the concentration of many heavy metals will be limited to levels below the environmentally possible maximum levels.

The preliminary examples that were used in this paper, to demonstrate the procedures, show that the concept allows to construct graphical reprentations formally similar to the traditional plots of the biological hazard potential vs. time. Contrary to the latter the new diagram presented here incorporates environmental effects as additional information.

In the present paper possible reductions of the high metal concentrations calculated for the pure systems Me-H_2O (as mentioned above or e.g. any kind of dilution) were consistently disregarded to arrive at results near the worst case. From Fig.6 one can take that then e.g. Bi activated by thermal reactions near the first wall will have to be isolated about 10^7 years, whereas Ni needs confinement for about 10^6 years, if one wants to avoid a potentially toxic concentration in drinking water.

On the other hand the property of the described method to deliver concentrations that are independent from the amounts of radionuclides having entered the environment and also from the metallic composition of the activated alloy is lost, as long as high concentrations prevail in the environment. Then the metals are no longer independent from one another.

The soil particles will certainly act as sinks for activated metals. That will have the positive effect of purifying the pore water. On the other hand it should be investigated, how much activity can accumulate in those sinks, and how stable they are.

Tab. 1 ACTIVITIES, BIOLOGICAL HAZARD POTENTIALS AND POTENTIAL HAZARD INDICES FOR 1 PPM-IMPURITIES IN FIRST WALL MATERIALS OF FUSION REACTOR

NEUTRON FLUX SPECTRUM: ONLY 14-MeV NEUTRONS CORRESPONDING TO 1 MW/M^2 OR
ONLY THERMAL NEUTRONS 4.5 x 10^{13} CM^{-2}S^{-1}

OPERATION TIME : 2 YEARS

mother(nx)daughter	ϱ_1	T_2	f_1	σ_{21}	λ_2	S_v $^{(1)}$	ALI$^{(2)}$	BHP$^{(3)}$	PHI$^{(4)}$ (P_i=1)	comment
1 σ_{21} 2	g/cm^3	a	%	barn	a^{-1}	mCi/cm^3	Ci/a	m^3/cm^3	cm^{-3}	
Li6(nα)H3	0.53	1.2+1 $^{(13)}$	7.5	940	5.61-2	5.1 -1	8-2	5.1-2	1.1-1	x3.4-1(5)
Be9(nγ)Be10	1.35	1.6+6	100	9.2-3	4.33-7	1.2-9	3-5°)	4.0-7	1.1-1	
C13(nγ)C14	1.7	5.7+3	1.1	9.0-4	1.2-4	2.3-10	8-3°)	2.9-10	2.9-7	
Cl35(nγ)Cl36	1.56	3.0+5	75.8	4.3+1	2.31-6	4.9-6	2-3	2.0-5	1.1+0	
Ca40(nγ)Ca41	1.55	1.3+5	96.9	4.0-1	5.33-6	1.2-7	2-3	4.7-7	1.1-2	
Ni58(nγ)Ni59	8.9	7.5+4	6.78+1	4.6+0	9.24-6	6.4-6	3-1	1.7-7	2.3-3	
Ni62(nγ)Ni63	8.9	1.0+2	3.7	1.42+1	6.93-3	7.6-4	1-2	6.1-4	1.1-2	
Se78(nγ)Se79	4.79	6.5+4	23.5	5.3-1	1.07-5	1.2-7	2-3	4.7-7	5.6-3	
Kr80(nγ)Kr81	2.6	2.1+5	2.25	1.4+1	3.3-6	4.9-8	-	-	-	
Zr92(nγ)Zr93	5.49	1.5+6	17.1	2.6-1	4.62-7	2.1-9	2-3	8.5-9	2.3-3	
Nb93(nγ)Nb94	8.4	2.0+4	100	1.2	3.47-5	5.5-6	8-4	5.5-5	2.0-1	
Mo92(nγ)Mo93	10.2	3.5+3	14.8	5.0-2	1.98-4	2.4-7	3-3	6.3-7	4.0-4	
Mo98(nγ)Tc99	10.2	2.1+5	24.4	1.3-1	3.3-6	1.6-8	3-3	4.2-8	1.6-3	
Ru96(nγ)Tc97	12.2	2.6+6	5.5	2.5-1	2.67-7	6.8-10	3-2	1.8-10	8.5-5	
Pd106(nγ)Pd107	12.0	6.5+6	27.3	2.9-1	1.07-7	1.4-9	3-2	3.7-10	4.4-4	
Ag107(nγ)Ag108*	10.5	1.3+2	51.8	3.0	5.46-3	1.2-3	8-4	1.2-2	2.8-1	
Cd112(nγ)Cd113*	8.65	1.5+1	24.0	6.0-2	4.75-2	7.5-5	2-5	3.0-2	8.1-2	
Sn120(nγ)Sn121*	7.3	5.0+1	32.8	1.0-3	1.4-2	4.0-7	2-3	1.6-6	1.5-5	
Te128(nγ)I129	6.24	1.6+7	31.8	2.0-1	4.42-8	2.0-10	5-6	3.1-7	9.0-1	
Xe134(nγ)Cs135	3.1	2.0+6	10.5	2.5-1	3.47-7	3.1-10	8-4	3.1-9	1.1-3	
Ba132(nγ)Ba133	3.5	1.1+1	9.5-2	8.5	6.6-2	2.5-0	2-3	7.9-5	1.6-4	
Sm144(nγ)Pm145	7.54	1.8+1	3.1	0.7	3.91-2	6.4-5	3-5°)	2.1-2	6.9-2	
Sm150(nγ)Sm151	7.54	9.3+1	7.4	1.02+2	7.45-3	4.1-3	1-2	3.3-3	5.5-2	x1.8-2(7)
Eu151(nγ)Eu152	5.26	1.24+1	47.8	5.9+3	5.59-2	8.0+0	1-3	6.4+1	1.4+2	x3.6-2(5)
Er162(nγ)Ho163	9.05	3.3+1	0.14	19.0	2.1-2	4.6-5	3-5°)	1.5-2	9.0-2	
Ho165(nγ)Ho166*	8.8	1.2+3	100	3.5	5.77-4	1.6-4	3-5°)	5.2-2	1.1+1	
Hf177(nγ)Hf178*	13.1	3.1+1	18.5	1.0-7	2.24-2	4.5-11	3-5°)	1.5-8	8.3-8	x6.2-1(5)(8
Re185(nγ)Re186*	21.0	2.0+5	37.4	1.12+2	3.47-6	2.4-5	2-3	9.6-5	3.5+0	
Ir191(nγ)Ir192*	22.5	2.41+2	37.4	9.24+2	2.88-3	1.8-1	1-3	1.4+0	5.9+1	x3.5-1(5) x5.7-2(6)
Bi209(nγ)Bi210*	9.8	3.5+6	100	1.4-2	1.98-7	1.9-10	8-4	1.9-9	1.2-3	

Tab. 1 cont.

mother(nx)daughter 1 σ_{21} 2	ρ_1 g/cm³	T_2 a	f_1 %	σ_{21} barn	λ_2 a^{-1}	S_v mCi/cm³	ALI Ci/a	BHP m³/cm³	PHI $(P_{13}=1)$ cm^{-3}	comment
Pb204(nγ)Pb205	11.4	1.4+7	1.4	6.61-1	4.95-8	3.7-11	3-3	1.0-10	2.5-4	
Pt192(nγ)Pt193	21.4	5.0+1	0.78	1.4+1	1.38-2	2.5-4	9-3°)	2.7-4	2.5-3	
Li7(n,αn)H3	0.53	1.2+1	92.5	3.2-1	5.61-2	1.9-3	8-2	1.9-4	4.1-4	
C13(nα)Be10	2.26	1.6+6	1.11	1.3-1	4.33-7	1.6-10	3-5°)	5.3-8	1.5-2	
N14(np)C14	0.81	5.7+3	99.6	3.6-2	1.2-4	3.7-7	8-3°)	4.6-7	4.7-4	
O17(nα)C14	1.14	5.7+3	3.9-2	5.3-2	1.2-4	2.5-10	8-3°)	3.1-10	3.2-7	
Al27(n2n)Al26	2.7	7.2+6	100	5.9-3	9.68-8	8.3-11	3-5°)	2.8-8	3.6-2	
K39(nα)Cl36	0.86	3.0+5	93.3	1.5-1	2.31-6	1.0-8	2-3	4.1-8	2.2-3	
Ca42(nα)Ar39	1.55	2.7+2	0.65	1.1-1	2.57-3	9.9-8	-	-	-	
K39(np)Ar39	0.86	2.7+2	93.3	3.5-1	2.57-3	2.7-5	-	-	-	
Ca42(n2n)Ca41	1.55	1.3+5	0.65	9.3-2	5.33-6	1.7-10	2-3	6.9-10	1.6-5	
Fe54(n2n)Mn53	7.86	3.7+6	5.8	3.6-1	1.87-7	8.3-10	5-2	1.3-10	8.9-5	
Ni60(n2n)Ni59	8.9	7.5+4	26.4	3.8-1	9.24-6	2.0-7	3-1	5.3-9	7.2-5	
Ni64(n2n)Ni63	8.9	1.0+2	0.95	1.1+0	6.93-3	1.5-5	1-2	1.2-5	2.1-4	
Sr88(nα)Kr85	2.6	1.1+1	82.6	2.0-1	6.44-2	4.5-4	-	-	-	
Zr94(n2n)Zr93	6.49	1.5+6	17.5	1.5+0	4.62-7	1.2-8	2-3	4.9-8	1.3-2	
Nb93(np)Zr93	8.4	1.5+6	100	4.5-2	4.62-7	2.7-9	2-3	1.1-8	3.0-3	
Mo92(n2n)Nb91	10.2	1.0+4	14.8	1.5-2	6.93-5	2.5-8	3-5°)	8.3-6	1.5-2	
Nb93(n2n)Nb92	8.4	1.0+8	100	4.1-1	6.93-9	3.7-10	3-5°)	1.2-7	2.2+0	
Mo92(np)Nb92	10.2	1.0+8	14.8	6.0-2	6.93-9	1.0-11	3-5°)	3.3-9	6.0-2	
Nb93(nn*)Nb93*	8.4	1.4+1	100	3.3-1	5.1-2	2.2-3	8-3	2.2-3	5.4-3	
Mo94(n2n)Mo93	10.2	3.5+3	9.1	5.6-1	1.98-4	1.6-6	3-3	4.3-6	2.7-3	
Ru98(n2n)Tc97	12.3	2.6+6	1.9	9.4-1	2.67-7	8.7-10	3-2	2.3-10	1.1-4	
Ru99(np)Tc99	12.3	2.1+5	12.7	1.5-2	2.67-7	1.1-9	3-3	3.0-9	1.2-4	
Cd114(n2n)Cd113*	8.65	1.46+1	28.8	8.6-1	4.75-2	1.3-3	2-5	5.2-1	1.4+0	
Sn122(n2n)Sn121*	7.3	5.0+1	4.7	9.0-1	1.38-2	5.1-5	2-3	2.0-4	1.8-3	
Te124(nα)Sn121*	6.24	5.0+1	4.6	3.2-3	1.38-2	1.5-7	2-3	6.0-7	5.4-6	
Te130(n2n)I129	6.24	1.57+7	34.5	1.2+0	4.42-8	1.2-9	5-6	2.0-6	5.8+0	
Ba137(np)Cs137	3.5	3.01+1	11.2	6.0-2	2.3-2	5.8-6	1-4	4.6-4	2.5-3	
Ba134(n2n)Ba133	3.5	1.05+1	2.4	8.5-1	6.6-2	5.1-5	2-3	2.1-4	3.9-4	
La138(n2n)La137	6.17	6.0+4	0.09	1.94+0	1.15-5	1.3-9	3-5°)	4.4-7	4.7-3	
Sm152(n2n)Sm151	7.54	9.3+1	26.7	2.07+0	7.45-3	3.0-4	1-2	2.4-4	4.0-3	x1.3-2(9)
Eu151(n2n)Eu150	5.26	3.5+1	47.8	6.1-1	1.98-2	2.9-4	3-5°)	9.8-2	6.2-1	x3.8-2(10)
Eu153(n2n)Eu152	5.26	1.24+1	52.2	7.6-1	5.59-2	1.1-3	1-3	8.9-3	2.0-2	x6.0-1(11)
Tb159(n2n)Tb158	8.27	1.5+2	100	2.6-1	4.62-3	9.1-5	3-5°)	3.0-2	8.2-1	
Er164(n2n)Ho163	9.05	3.3+1	1.6	2.11+0	2.10-2	5.7-5	3-5°)	1.9-2	1.1-1	
Ir193(n2n*)Ir192*	22.5	2.4+2	62.6	5.9-1	2.87-3	1.8-4	1-3	1.5-3	6.3-2	x2.5-1(12)
Pt194(n2n)Pt193	21.4	5.0+1	32.9	1.16+0	1.39-2	8.5-4	9-3°)	9.4-4	8.5-3	
Pb206(n2n)Pb205	11.4	1.4+7	24.1	2.44+0	4.95-8	2.3-9	3-3	6.3-9	1.6-2	
Bi209(n2n)Bi208	9.8	3.7+5	100	2.48+0	1.88-6	3.2-7	3-5°)	1.1-4	7.1+0	

Comment to Tab. 1

(1) S_V = specific activity at shut-down (Ci/cm^3) =
$S_V = n_1 R_{21} (1- \exp(- \lambda_2 t))/c_1$
$c_1 = 3.16 \times 10^7 \times 3.7 \times 10^{10}$
n_1 = number of mother atoms per cm^3 = $\rho_1 f_1 N_L 10^{-6}/M_1$
ρ_1 = density of mother element (g/cm^3)
f_1 = fraction of mother nuclide present in natural element
N_L = Avogadro's number (6×10^{23} particles per mole)
M_1 = molar mass of mother nuclide (g/mole)
R_{21} = reaction rate (a^{-1}) = $\emptyset \sigma_{21}$
$\emptyset = 4.5 \times 10^{13}$ cm^{-2}s^{-1} (either thermal or 14-MeV neutrons, the latter correspond to 1 MW/m^2 first wall load)
R_2 = sum of reaction rates of daughter (a^{-1})
R_1 = sum of reaction rates of mother (a^{-1})

(2) ALI = annual limit for intakes of radionuclides by workers, after Adams, N. et al. NRPB-R82 (1978), ICRP-Publ.30, Ann.ICRP 2 (1979) and 3 (1980), values marked by ^{e}) are MPC (maximum permissible concentrations in water for occupational exposure (168 hours per week) after ICRP-Publ.2 (1959))

(3) BHP = biological hazard potential (m^3 water/cm^3 of impure material) = $S_V/(0.1\text{ALI}/0.8)$ or = $S_V/(0.1\text{MPC})$
0.1 = reduction factor to adjust occupational exposure to exposure of public
0.8 = annual amount of drinking water (m^3)

(4) PHI = potential hazard index(for probability of waste nuclide i entering man = 1) = $(S_V/\text{ALI})(T/\ln 2)$ = BHP $(0.1/0.8)/(T/\ln 2)$
T = half-life of radionuclide

(5)-(12) reduction factors due to transmutation of mother(1) and daughter (2). If only σ_1 is given in corresponding comment (see below) transmutation of mother is considered only.

(6) $\sigma_1^{n\gamma}$ = 924 barn, $\sigma_2^{n\gamma}$ = 1100 barn
(7) $\sigma_1^{n\gamma}$ = 102 barn, $\sigma_2^{n\gamma} = 1.5 \times 10^4$ barn
(8) $\sigma_1^{n\gamma}$ = 366 barn

Comment to Tab. 1 cont.

(9)-(12) reduction factors applicable only, if 14-MeV and thermal neutron fluxes are present simultaneously

(9) $\sigma_1^{n\gamma}$ = 206 barn, $\sigma_2^{n\gamma}$ = 15000 barn
(10) $\sigma_1^{n\gamma}$ = 9200 barn
(11) $\sigma_1^{n\gamma}$ = 390 barn
(12) $\sigma_1^{n\gamma}$ = 110 barn, $\sigma_2^{n\gamma}$ = 1100 barn

(13) 1.2 + 1 stands for 1.2 x 10^1

reduction factor = N_{trans}/N_o, $N_{trans} = \frac{n_1 R_{21}}{R_2+\lambda_2-R_1} (\exp(-R_1 t)-\exp(-(R_2+\lambda_2)t$

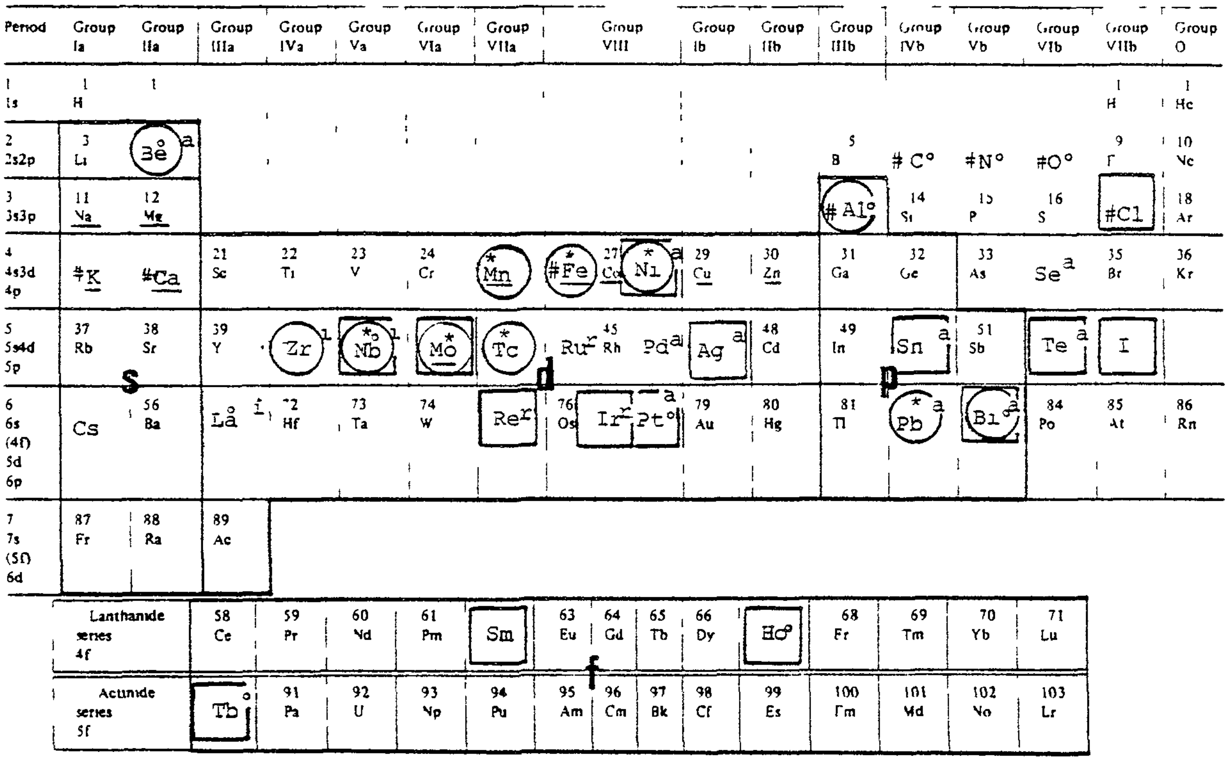
$$N_o = n_1 R_{21}(1-\exp(-\lambda_2 t))/\lambda_2$$

t = 2 years

Meaning of PHI

Intake of ALI causes a total committed dose equivalent in body of less than 5 rem for stochastic fatal cancer (ICRP-Publ.26, ICRP-Publ.30, Pt.1). The ALIs are computed, using two values U_s and SEE, U_s being the number of transmutations of radionuclide in source tissue over a period of 50 years after intake. U_s is calculated by use of a computerized metabolic model based on first order linear differential equations describing the retention in tissues. SEE is the specific effective energy for the radiation from radionuclide (modified by quality factor) absorbed in tissue, when the radionuclide has undergone transmutation in any source tissue (source- and target-tissues are selected organs). Both U_s and SEE are tabulated in ICRP-Publ.30, Suppl.

Tab. 2 Periodic table (after Cotton and Wilkinson, Basic Inorganic Chemistry, Wiley (1976)) showing radionuclides of Fig. 1 with half-life ≥ 50 a



Period	Group Ia	Group IIa	Group IIIa	Group IVa	Group Va	Group VIa	Group VIIa	Group VIII			Group Ib	Group IIb	Group IIIb	Group IVb	Group Vb	Group VIb	Group VIIb	Group 0
1 1s	1 H	1															1 H	1 He
2 2s2p	3 Li	Be° a											5 B	# C°	#N°	#O°	9 F	10 Ne
3 3s3p	11 Na	12 Mg											#Al°	14 Si	15 P	16 S	#Cl	18 Ar
4 4s3d 4p	#K	#Ca	21 Sc	22 Ti	23 V	24 Cr	*Mn	*#Fe	27 Co	*Ni a	29 Cu	30 Zn	31 Ga	32 Ge	33 As	Se a	35 Br	36 Kr
5 5s4d 5p	37 Rb	38 Sr	39 Y	Zr i	*°Nb i	*Mo	*Tc	Ru r	45 Rh	Pd a	Ag a	48 Cd	49 In	Sn a	51 Sb	Te a	I	
6 6s (4f) 5d 6p	Cs	56 Ba	La° i	72 Hf	73 Ta	74 W	Re r	76 Os	Ir r	Pt° a	79 Au	80 Hg	81 Tl	*Pb a	Bi° a	84 Po	85 At	86 Rn
7 7s (5f) 6d	87 Fr	88 Ra	89 Ac															

Lanthanide series 4f	58 Ce	59 Pr	60 Nd	61 Pm	Sm	63 Eu	64 Gd	65 Tb	66 Dy	Ho°	68 Er	69 Tm	70 Yb	71 Lu
Actinide series 5f	Tb°	91 Pa	92 U	93 Np	94 Pu	95 Am	96 Cm	97 Bk	98 Cf	99 Es	100 Fm	101 Md	102 No	103 Lr

Comment

Mn	(big letters, typewriter): Those radionuclides have a BHP bigger than 10^{-11} m^3/cm^3,
☐	BHP bigger than 10^{-6} m^3/cm^3 for either fast or thermal neutron reaction
○	potential structural alloy constituent or neutron multiplier with BHP bigger than 10^{-11} m^3/cm^3
°	BHP calculated using MPC
–	essential metals (Vahrenkamp, 1973)
#	considered non-toxic (Wood, 1974)
*	relevant for INTOR-(316SS)-structure or neutron multiplier
i	very insoluble (Wood, 1974)
r	very rare in earth's crust (concentration lower than 0.1 ppm)
a	relatively accessible and very toxic (Wood, 1974)

Tab. 3 Predominant heavy metal species under all conditions of computations in the models of Morel et al. (1973). The species of each metal are listed according to pH (after Förstner, Wittmann, 1979)

	Species accounting for more than 90 %	...for a few percent
Fe	$Fe(OH)_2^+$, $FePO_{4s}$, $Fe(OH)_{3s}$, $FeCO_{3s}$ FeS_s, $FeSiO_{3s}$	
Mn	Mn^{++}, $MnCO_{3s}$, MnO_{2s}, MnS_s	$MnHCO_3^+$, $MnSO_4$, $MnCl^+$
Ni	Ni^{++}, $Ni(OH)_{2s}$, NiS_s	$NiSO_4$
Pb	Pb^{++}, $PbCO_{3s}$, PbO_{2s}, PbS_s	$PbSO_4$, $PbCl^+$

Tab. 4 Potential concentrations and reference concentrations of activated materials (metals and actinides) in aqueous solutions

mother(nx)daughter (T) a	S_v (1) mCi/cm³	S_v/ρ mCi/g	c (2) mol/l	c* (3) Ci/m³	c_w^* (4) Ci/m³	c^*/c_w^*
Be 9(nγ)Be10(2+6)	1.2-3	6.3-4	<1-4	<6.3-7	3-6$^{o)}$	<2-1
C 13(no)Be10(2+6)	1.6-4	8.4-5	<1-4	<8.4-8	3-6$^{o)}$	<3-2
Al27(n2n)Al26(7+6)	8.3-5	3.1-5	<1-2	<8.0-6	3-6$^{o)}$	<3+0
Fe54(n2n)Mn53(4+6)	8.3-4	1.1-4	1	5.9-3	6-3	1
Ni58(nγ) Ni59(8+4)	6.4+0	7.2-1	1	4.2+1	4-2	1+3
Ni60(n2n)Ni59(8+4)	2.0-1	2.2-2	1	1.3+0	4-2	3+1
Ni62(nγ) Ni63(1+2)	7.6+2	8.5+1	1	5.4+3	1-3	5+6
Ni64(n2n)Ni63(1+2)	1.5+1	1.7+0	1	1.1+2	1-3	1+5
Nb93(np) Zr93(2+6)	2.7-3	4.2-4	<1-4	<3.9-6	3-4	<1-2
Zr94(n2n)Zr93(2+6)	1.2-2	1.8-3	<1-4	<1.7-5	3-4	<6-2
Zr92(nγ) Zr93(2+6)	2.1-3	3.2-4	<1-4	<3.0-6	3-4	<1-2
Mo92(nγ) Mo93(4+3)	2.4-1	2.4-2	1	2.2+0	4-4	5+3
Mo94(n2n)Mo93(4+3)	1.6+0	1.6-1	1	1.5+1	4-4	4+4
Mo92(n2n)Nb91(1+4)	2.5-2	-	(5)	-	-	-
Mo98(nγ) Tc99(2+5)	1.6-2	1.4-3	1	1.4-1	4-4	3+2
Pb204(nγ)Pb205(1+7)	3.7-5	3.3-6	1	6.7-4	4-4	2+0
Pb206(n2n)Pb205(1+7)	2.3-3	2.0-4	1	4.2-2	4-4	1+2
Bi209(n2n)Bi208(4+5)	3.2-1	3.3-2	1	6.8+0	3-6$^{o)}$	2+6
Bi209(nγ) Bi210(4+6)	1.9-4	1.9-5	1	4.1-3	1-4	4+1
Am241 (4+2)		3.4+3 (6)	1 (7)	8.1+5	2-7	4+12
Am243 (7+3)		1.9-1	1	4.7+4	2-7	2+11
Pu239 (2+4)		6.1-2	1	1.5+4	7-7	2+10
Np237 (2+6)		7.7-4	1	1.8+2	1-8	2+10

Comment to Tab. 4

(1) S_v = activity for material being composed of the pure mother element, e.g. in first row molybdenum. S_v in this table is equal to $10^6 x S_v$ of Tab.1

(2) c = maximum concentration in aqueous solutions at atmospheric pressure, 25 C and $2.8 < pH < 11$, in the absence of complexing agents, after Pourbaix, M. "Atlas of electrochemical equilibria", Pergamon (1966). If compound can become soluble, c = 1 is assumed .

(3) c* = potential concentration in drinking water, $c^* = S_v Mc/\rho$
M = molar mass of daughter
S_v/ρ = "specific activity" of activated material (Ci/g of mother element)

(4) c_w^* = concentration of radionuclide in drinking water leading to an exposure of 0.5 rem/a, $c_w^* = 0.1/0.8 ALI$, values marked by $^{o})$ are 0.1MPC instead of 0.1/0.8 ALI

(5) unsufficient data, for all pH Nb tends to cover by oxide film, the persistence of which is unsufficiently known (see Pourbaix p.248)

(6) specific activities after Gasteiger, R. KFK 2373 (1976)

(7) solubilities after Ames, L.L et al. BNWL-1983 (1976), oxidation states: Am^{3+}, Pu^{4+}, Np^{5+}

Tab. 5 Carrier substances and mechanisms of heavy metal bonding (from Förstner and Patchineelam, 1976)

heavy metal - hydroxides - carbonates - sulfides	..... (1)	Precipitation as a result of exceeding the solubility product
minerals of natural rock debris, detritus heavy minerals etc.	..... (2)	metal bonding predominantly in inert positions
(hydr)oxides of Fe and Mn	..... (3)	(a) physico-sorption (b) chemical sorption (exchange of H^+ in fixed positions) (c) coprecipitation as a result of exceeding the solubility product
calcium carbonate	..... (4)	(a) physico-sorption (b) pseudomorphosis (c) coprecipitation as a result of exceeding the solubility product
bitumen, lipids, humic substances, residual organics	..... (5)	(a) physico-sorption (b) chemical sorption (exchange of H^+ in COOH, OH-groups) (c) complexes

Tab. 6 Total metal concentration (in ppm, Fe in %) of the sediment samples, to which chemical associations are presented in Fig. 8 (from Förstner, Wittmann, 1979)

	Mn	Fe	Ni	Co	Cr	Cu	Pb	Zn	Cd
Sea of Okhotsk (Nissenbaum, 1972)	575	5.5%	79	--	--	44	--	98	--
Yukon River (Gibbs, 1973, 1977)	1270	6.3%	136	40	115	416	--	--	--
Dead Sea (Nissenbaum, 1974)	638	1.9%	46	23	--	43	13	52	12
Los Angelos Hr. (Gupta and Chen, 1975)	493	4.5%	47	--	178	568	332	612	2
Lower Rhine (Patchineelam, 1975)	750	3.7%	167	35	397	376	333	1096	28

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Patchineelam, S.R.: Untersuchungen über die Hauptbindungsarten und die Mobilisierbarkeit von Schwermetallen in fluviatilen Sedimenten. Diss. Univ. Heidelberg, 136 pp. (1975)

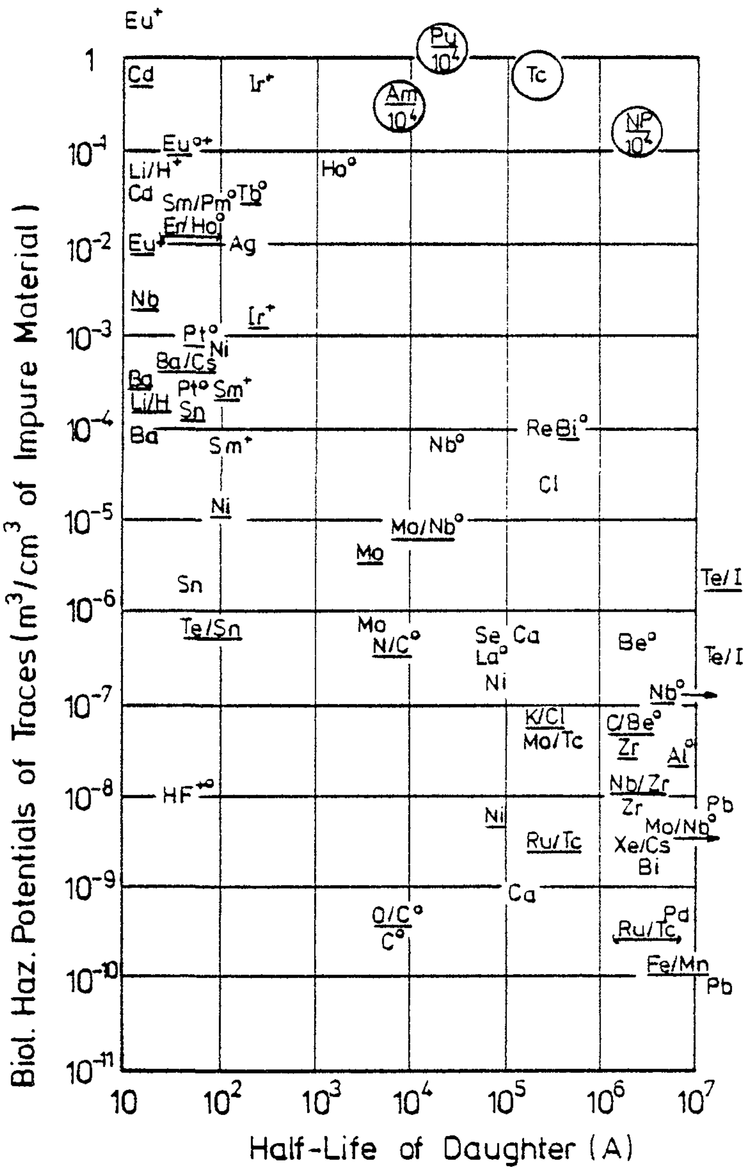


Fig. 1 Biological hazard potential of traces (1 ppm) in the fusion reactor first wall
irradiation conditions: $\emptyset = 4.5 \times 10^{13}$ $cm^{-2}s^{-1}$ either thermal or
14 MeV neutrons
$t = 2$ years

chemical symbol gives mother and (if different behind /) daughter.
activation by fast neutrons is underlined

Symbols: ° : old MPC-values have been used
\+ : reduction factors not incorporated in this figure
(for reduction factors see comment (9)-(12) in Tab.1)
o : symbols in circles refer to spent LWR-fuel

Note that about 3 cm^3 of first wall corresponds to 1 kWt and about 1 cm^3 of LWR-fuel corresponds to the same thermal power.

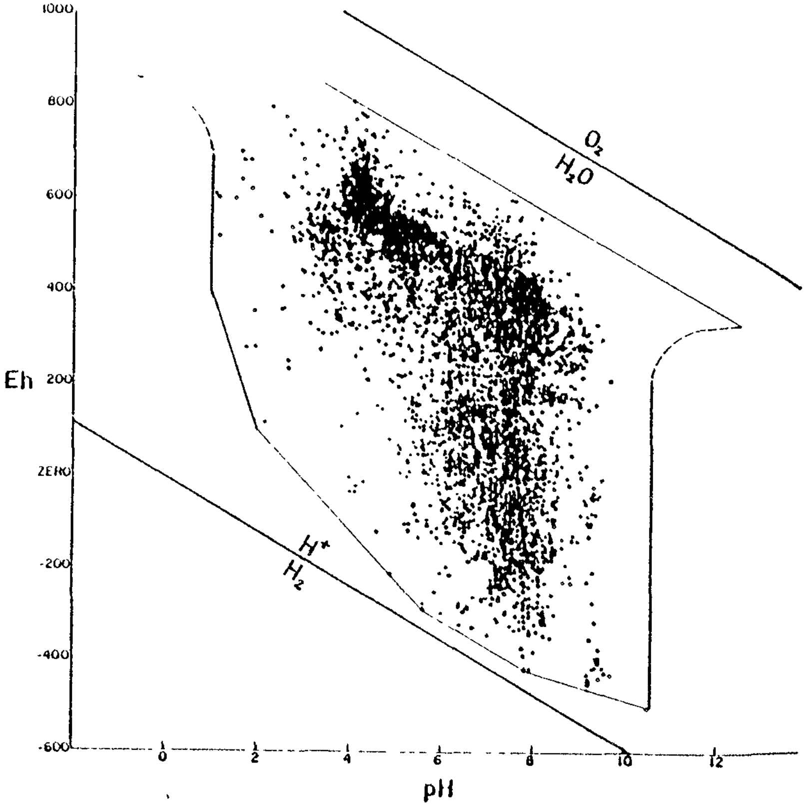


Fig. 2 Distribution of E_H-pH measurements of natural aqueous environments (after Baas Becking et al., 1960)

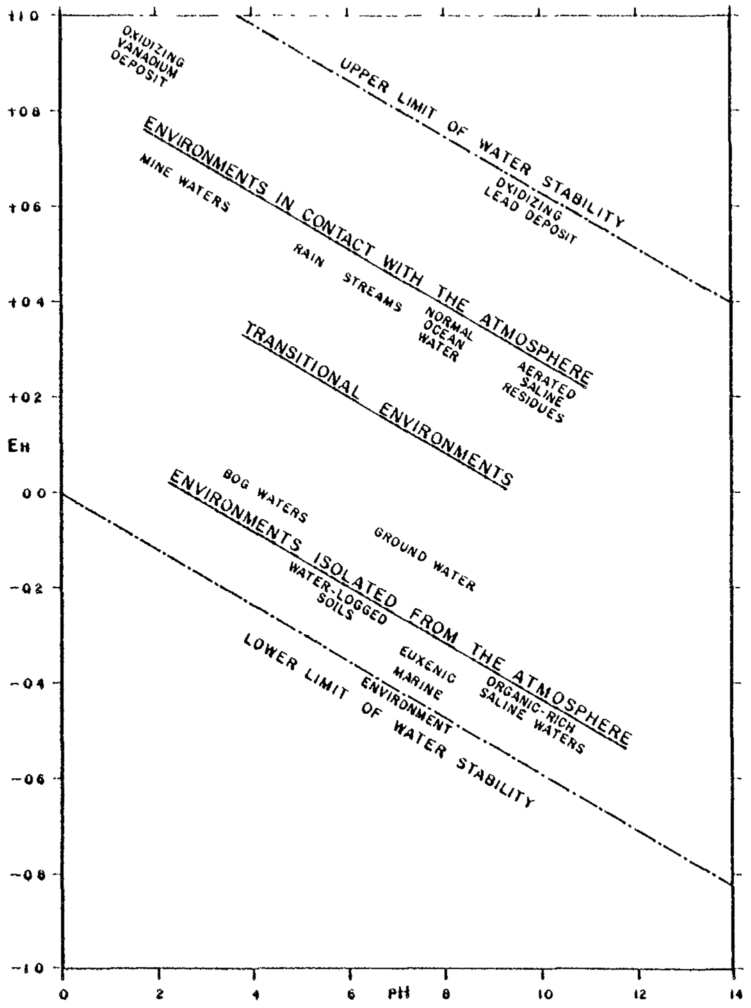


Fig. 3 Approximate position of some natural environments as characterized by E_H-pH (after Garrels and Christ, 1965)

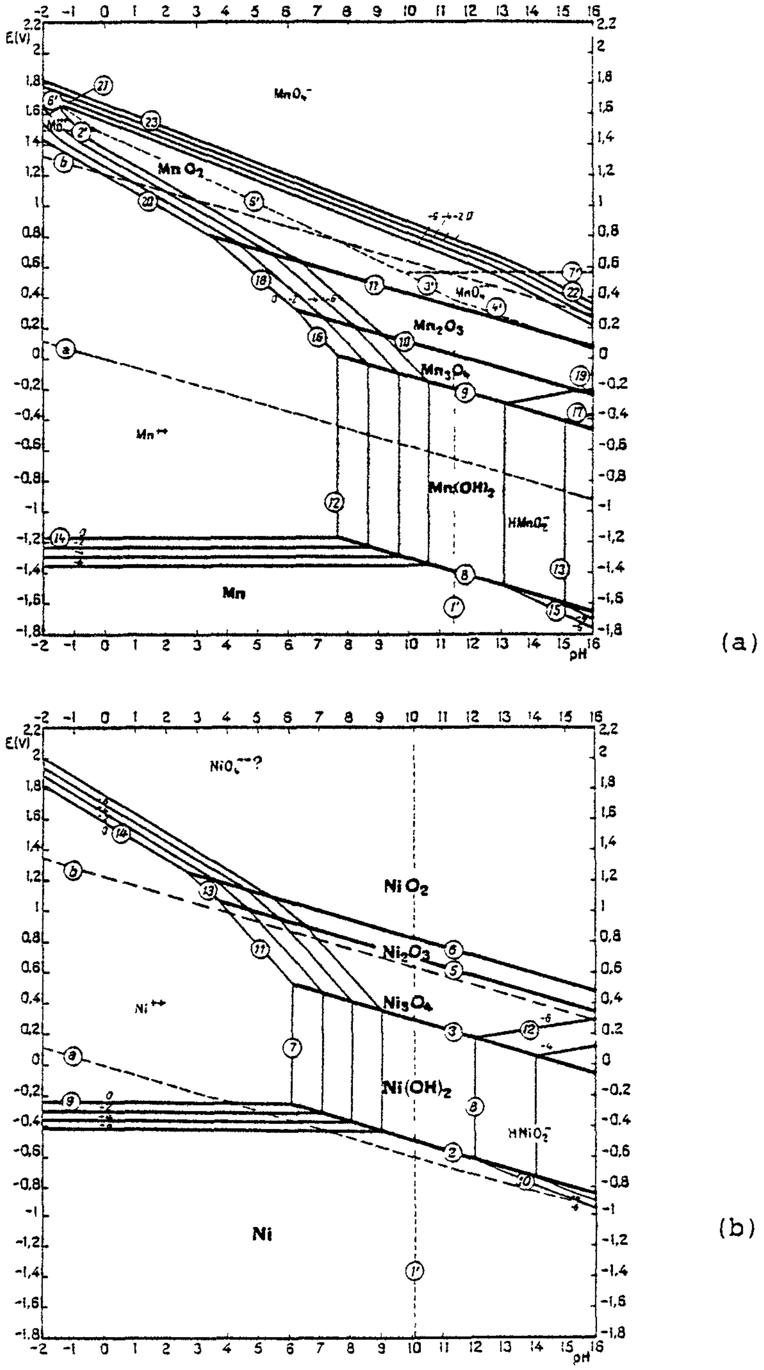


Fig. 4 E_H-pH equilibrium diagram for the systems (a) Mn-H_2O, (b) Ni-H_2O at 25°C and 1 atm (after Pourbaix, 1966)

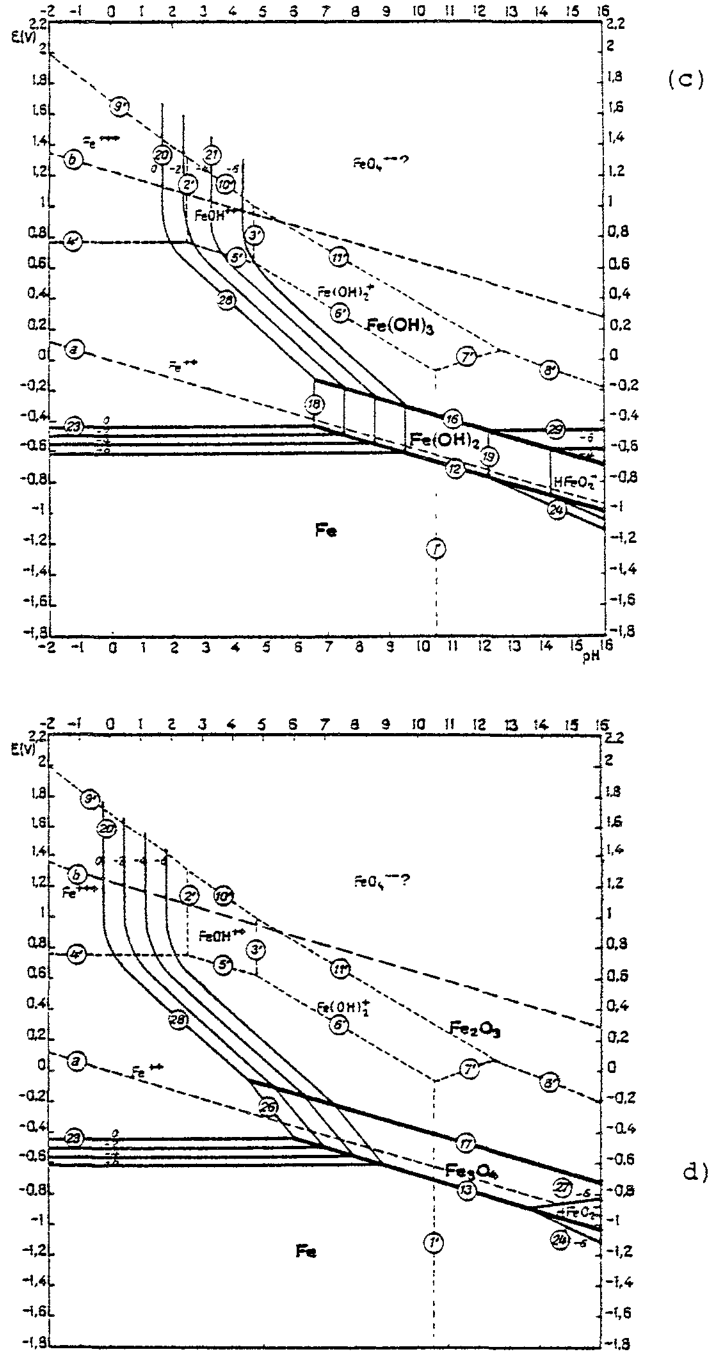


Fig. 4 E_H-pH equilibrium diagram for the systems

(c) $Fe-H_2O$ (considering as solid substances only Fe, $Fe(OH)_2$ and $Fe(OH)_3$

(d) $Fe-H_2O$ (considering as solid substances only Fe, Fe_2O_3 and Fe_3O_4

at 25°C and 1 atm (after Pourbaix, 1966)

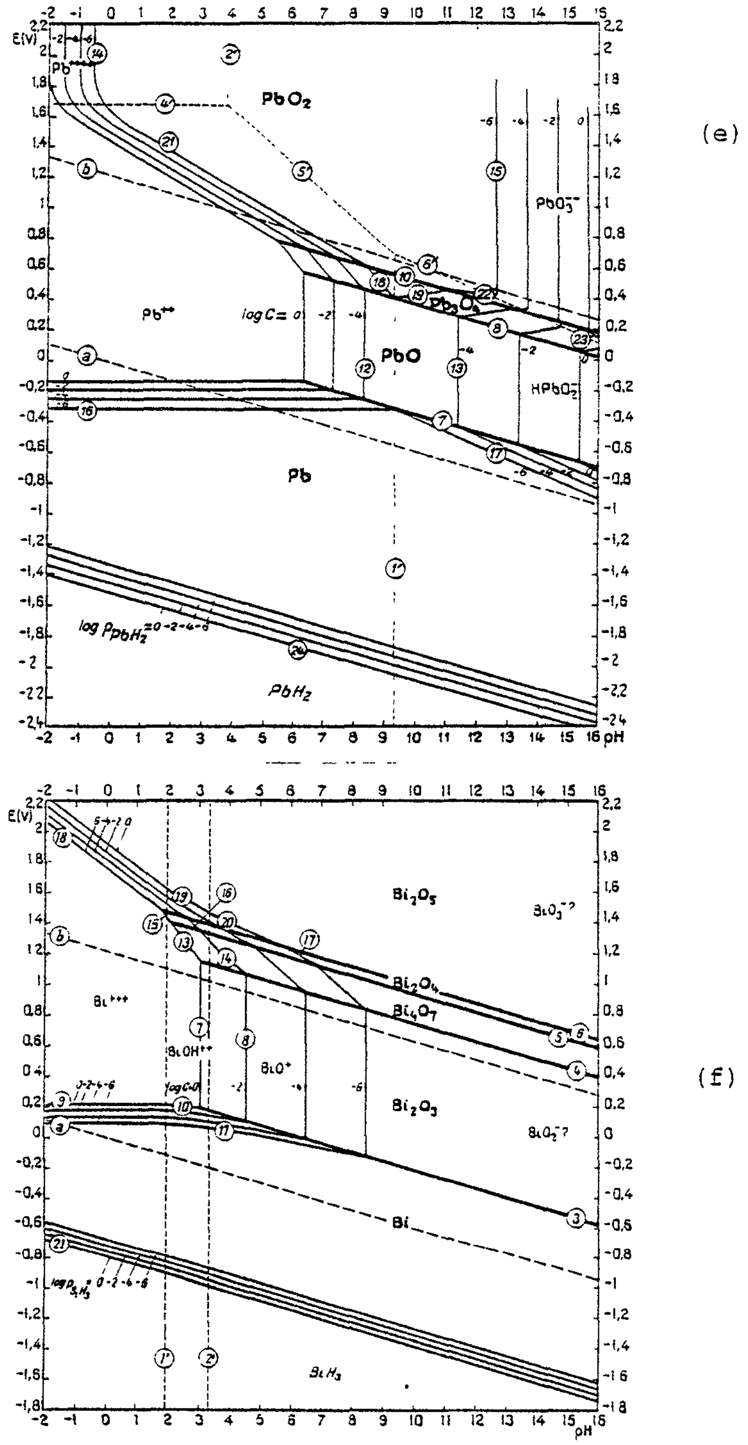


Fig. 4 E_H-pH equilibrium diagram for the systems (e) $Pb-H_2O$, (f) $Bi-H_2O$ at 25°C and 1 atm (after Pourbaix, 1966)

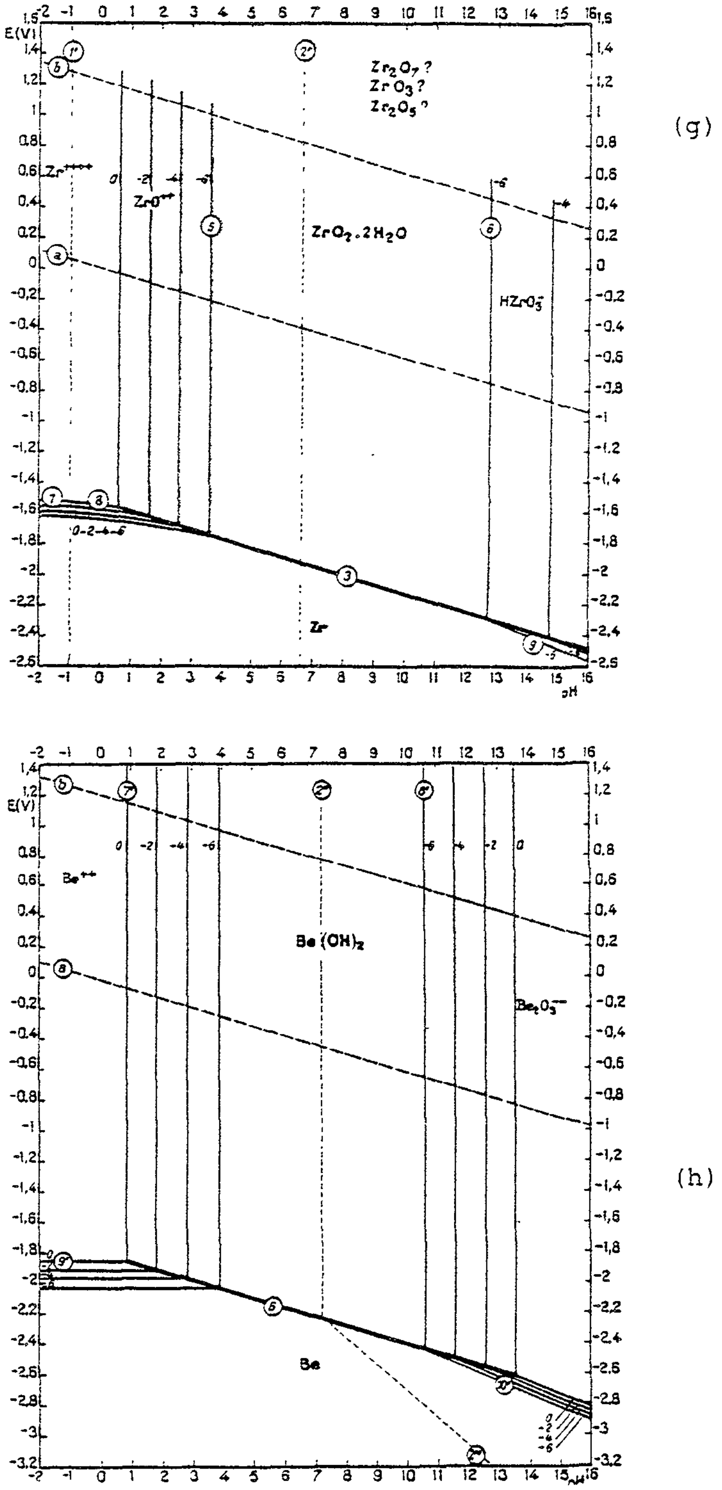


Fig. 4 E_H-pH equilibrium diagram for the systems (g) $Zr-H_2O$, (h) $Be-H_2O$ at 25°C and 1 atm (after Pourbaix, 1966)

(i)

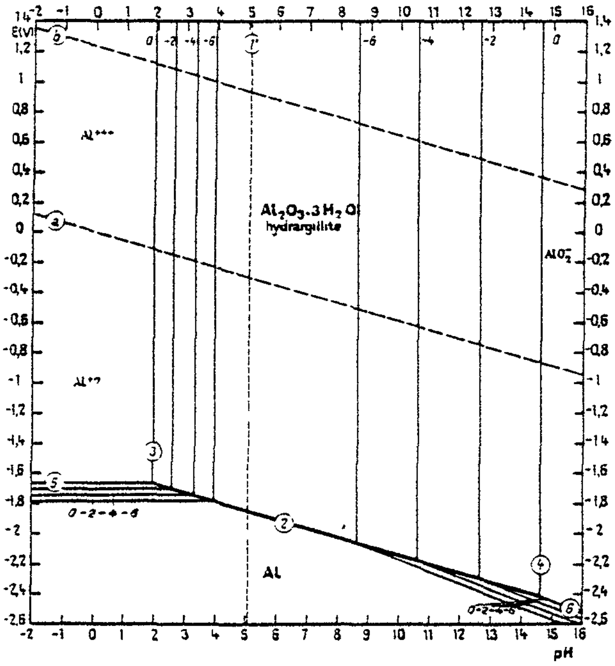
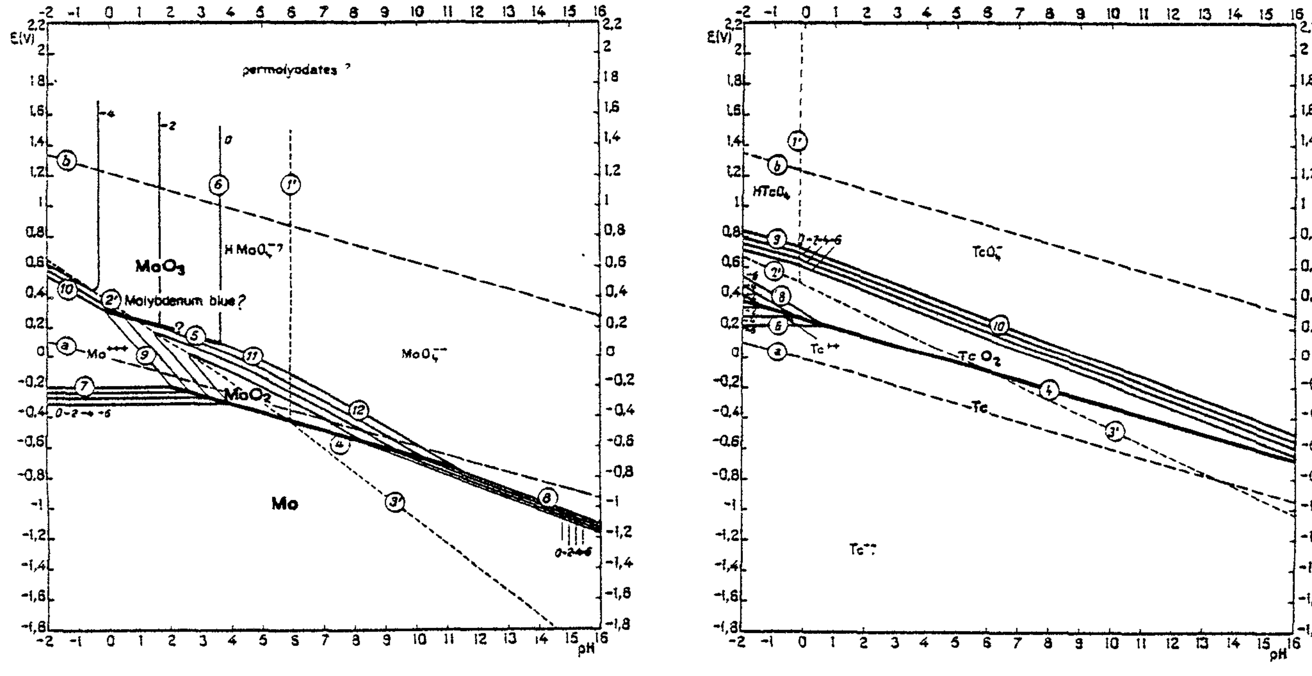


Fig. 4 E_H-pH equilibrium diagram for the system $Al-H_2O$ at 25°C and 1 atm (after Pourbaix, 1966)



(k) (l)

Fig. 4 E_H-pH equilibrium diagram for the systems (k) $Mo-H_2O$, (l) $Tc-H_2O$ at 25°C and 1 atm (after Pourbaix, 1966)

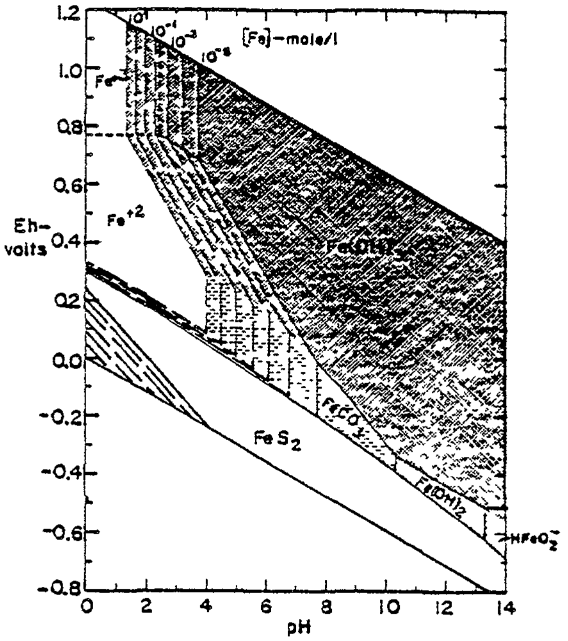


Fig. 5 Solubility of Fe in relation to pH and E_H at 25°C and 1 atm. Total dissolved sulfur 10^{-4} moles/l, bicarbonate species 10^{-3} moles/l (after Hem, 1965)

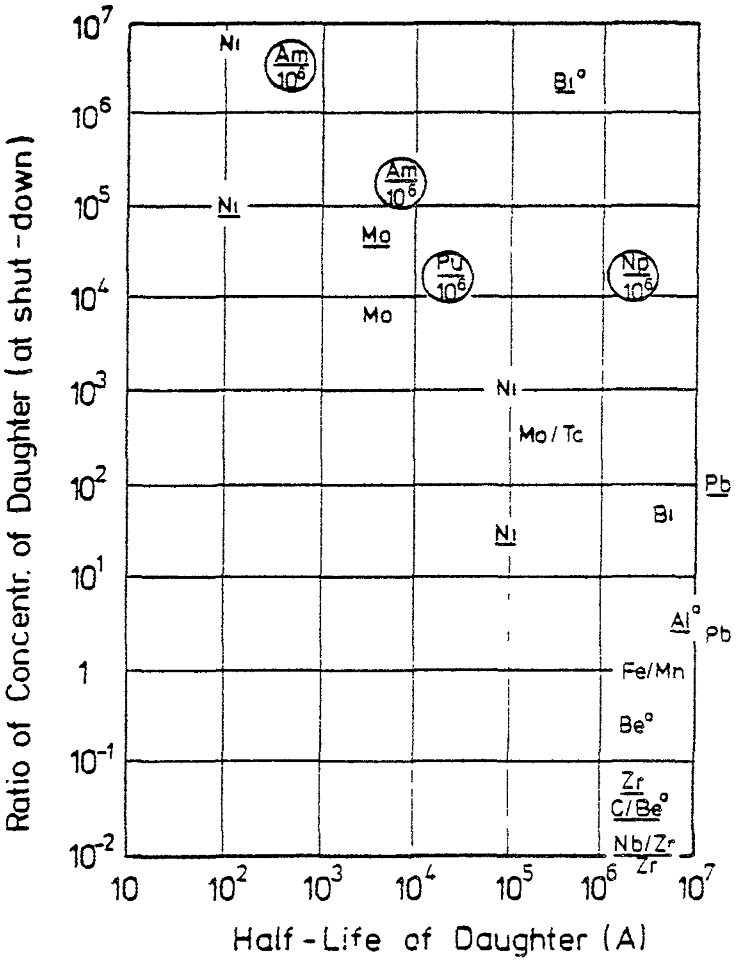


Fig. 6 Ratio of potential concentrations and reference concentrations (irradiation conditions and symbols as explained in Fig. 1)

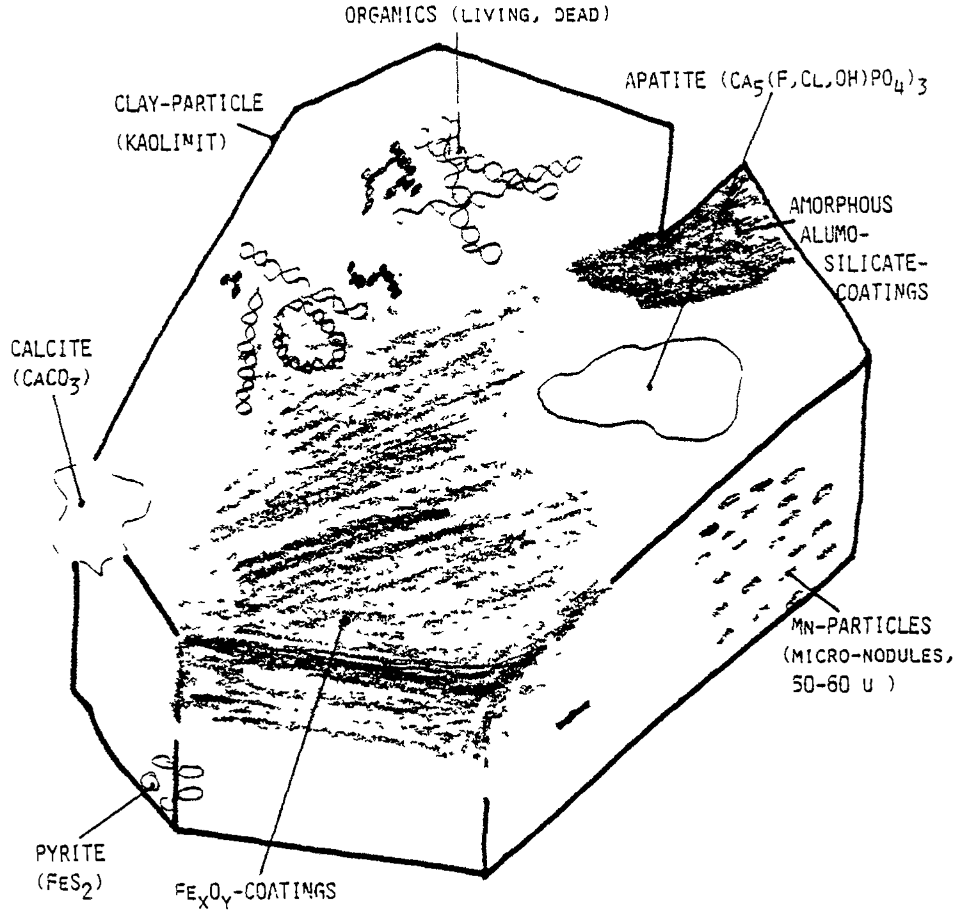


Fig. 7 Adsorption sites for transmission metals on clay (after E. Jenne, 1978)

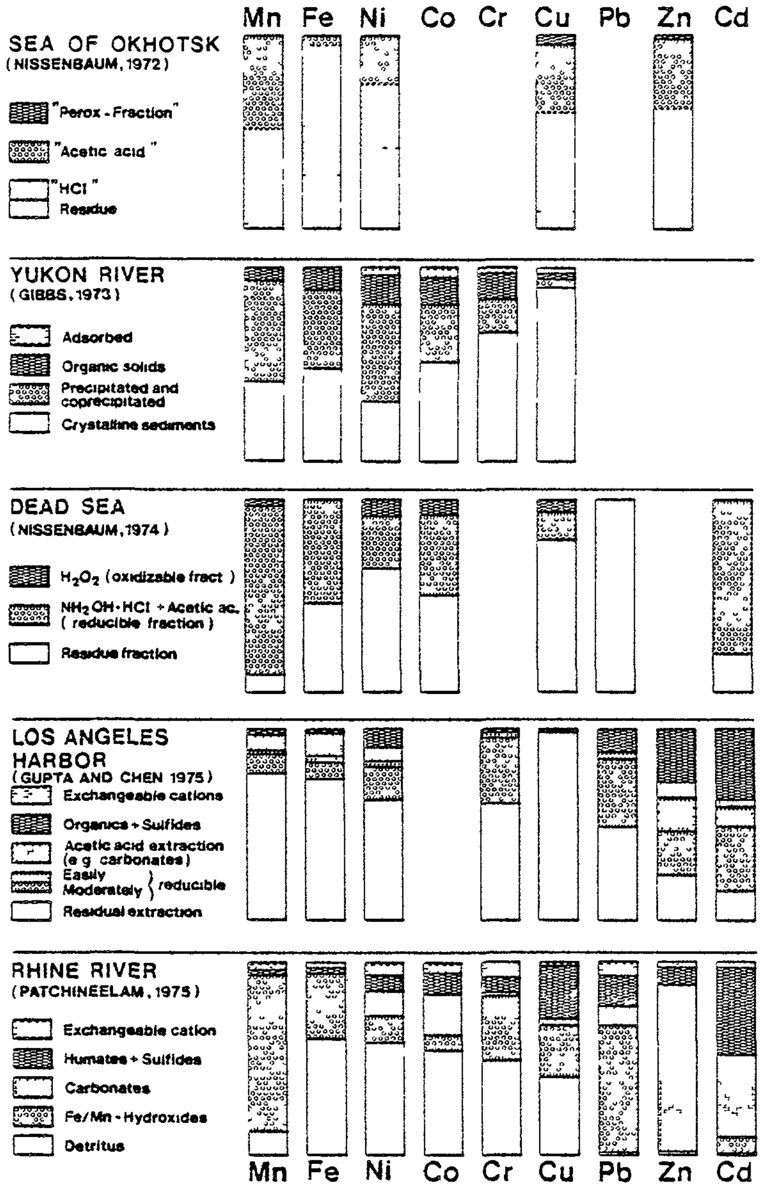


Fig. 8 Chemical associations of heavy metals in sediments from natural and polluted aquatic environments (after Förstner and Wittmann, 1979) (Further details see Tab. 6.)

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DISCUSSION

J.G.CROCKER: You considered LWRs. Why did you not make a comparison with the breeder?

J.GRUBER: It is just because of a historical reason; firstly I found more literature on LWRs and secondly I did all the calculations for the LWRs while being in the project leadership of the German risk analysis on fission reactor waste management.

J.G.CROCKER: If you had made the comparison for the breeder, would not fusion, by comparison, have come out better?

J.GRUBER: Not much. I have looked at the plutonium generated in the fuel elements of a LWR where you have a conversion factor (number of Pu generated per fission) of about 0.3 - 0.4. In the blanket of a breeder reactor you have a conversion factor of 1.4. The factor of 3 or 4 is not important for waste calculations.

J.C.CHICKEN: How did you come to make this study?

J.GRUBER: While organizing the scientific work in the fission reactor waste study I came in contact with the models that are used there and I transferred those models into fusion.

J.C.CHICKEN: That was a waste study commissioned in Germany?

J.GRUBER: Yes, a study that involved roughly 10 Million DM.

J.G.CROCKER: The fact that the distribution of pH of our environment is changing as a function of time is something that one has not thought about before. You showed the lakes becoming much more acetic than they were 50 years ago, which could make a significant difference in the way in which your models show how these radionuclides will migrate in the enviornment. This in turn could change up-take, distribution in man, animals etc.

J.GRUBER: Radioecologists only now start to consider this pH dependency. Few years ago they spoke about one value to characterize the concentration in the solution, the distribution of radionuclides between soil and soil solution. Now they say that one has to introduce at least a pH dependent value.

J.SCHNEIDER: You classified the hazard potential, the potential hazard index, into short term and long term and you said that the long term hazard potential or potential hazard index for fission and fusion is comparable when comparing the actinides with the activated metals. Can you say something on the short term hazard potential?

J.GRUBER: No, unfortunately I have no such numbers with me. The short term hazard is not relevant for waste considerations. I have the feeling that isolation of the radionuclides over a time span of 100 years is not a problem. This can be achieved by concrete tombs for example.

J.G.CROCKER: At least the studies that I am familiar with show that, typically, fusion is down in total BHP by 2-3 orders of magnitude in the early times. But even later on, when you start adding-up the total BHP you start to get down to the values that are typical of coal ash or something like that. So there may be a comparison of fission with fusion, but wastes start to decay to the point where there are other things that we fairly commonly accept, that have a very high BHP compared to these or at least comparable BHPs.

M.S.KAZIMI: I wonder whether you could point out where the BHP value of natural uranium could be in your graphs and at what time after shut down the decay would become comparable to its BHP value.

J.GRUBER: The comparison with natural uranium is, in my opinion, not proper to fusion, and this is the reason why I left it out. However, it is used for fission reactors where on starts-up with uranium, does something with it and puts it back into the environment. Then one can make a comparison. When one considers a uranium ore containing 5% pitch-blende then one has about 2-3 orders of magnitude between the BHP of the long lived actinides and an equal amount of uranium (weight). However one should not only compare the biological aspect but also the environmental aspect. I should probably include the uranium level in my concentration plot. It would appear there at $c^*/c^*_w = 9 \times 10^4$.

M.S.KAZIMI: I think it should be emphasized in the waste disposal assessment that there are a number of steps that people normally do not take into account when making a statement about potential for hazards (except as for certain studies that perhaps B.Cohen does at the University of Pittsburg). Not only does one take a BHP at the particular point but also attempt to take into consideration the time it will take for any water that may get into a site to resurface and hence get into the food chains. And if one combines all these things together one finds that the total probability, if you wish, for intake of some of the waste isotopes becomes quite low. I think that in fusion we were looking for the potential for recycling some of the structures that have been used somewhere along the line perhaps at 50-100 years after their initial use and therefore it is not possible for us immediately to say that we will resort to the same burial techniques that the fission people have considered.

J.GRUBER: The aim of my calculations was to find out the answer to the following question: Do we need a long-term repository like a geological formation that has been isolated from the biosphere for e.g. million years? What happens in the worst case if one releases dust, aerosols of niobium from reprocessed fusion material. Will the dust be fixed so tightly to soils that it can be neglected? And that is what can be expected from what the soil scientists tell us.

M.S.KAZIMI: I am concluding from the current information you have that recycling does not seem feasible because of the high BHP. Is this correct?

J.GRUBER: The information I have points towards that it is feasible.

J.L.ANDERSON: I understand from the graphs that, at some point after storage,the hazard is quite small for these that are below ratio 1. If zirconium and niobium dust was released then it would be so tightly down in the soil that the track would be very small. Is that the correct interpretation?

J.GRUBER: Yes.

J.L.ANDERSON: So this encourages consideration of reprocessing.

J.GRUBER: Yes. What I consider in Fig.6 or Tab.4 is the concentration of a radionuclide. Of course, what matters also is the amount of contamination that has entered the environment: A small grain or particle of e.g. nickel-hydroxide will contaminate only a very limited area of soil or volume V of groundwater, regardless how high the potential concentration will be. It is important to know, how large a volume will be contaminated by a potentially radiotoxic radionuclide (ratio $c^*/c^*_w > 1$) in Fig.6). Whenever the volume V is small as compared to the annual consumption, the dilution of contamination with uncontaminated water has to be taken into account. The amount V of potentially contaminated water can be calculated based on the BHP (Fig.1 or Tab.4): V is the BHP divided by the ratio c^*/c^*_w of Fig.6.

J.G.CROCKER: We had a lengthy discussion about your paper. However, some of us did not get quite the same message and we want to make sure that we are talking on the same wavelength. Could you give us just a short dissertation on your conclusions again without the confusing factor of the graphs which at least I found difficult to track. You have done a lot of interesting work and the approach was somewhat new to some of us. We have to fully understand this approach.

J.GRUBER: I wanted to emphasize that the normal characterization of the waste by the BHP can be improved. This characterization by the BHP takes only the metabolism of man into account. Using the BHP we can answer the question about which radionuclide intake we want least and which radionuclide intake we want to avoid. Now, if a quite hazardous radionuclide does not at all enter the food chain, its existence can be neglected because it does not affect us. What governs the radionuclide's entrance via the drinking water is the concentration of that radionuclide in the soil solution, in the groundwater. I used those graphs (Fig.4) to show the potential solubilities of the radionuclides in soil. The graphs demonstrate that a large number of heavy metals is readily soluble in soil. Consequently the potential contamination of the groundwater due to many heavy metal radionuclides is higher than 0.1MPC. And in Fig.6 I plotted for each radionuclide the ratio of the potential contamination and 0.1MPC versus its half life. The necessary confinement time for those unpleasant radionuclides would then be the time, during which their specific activities (measured in Ci per g of radionuclide) have decreased sufficiently and the contamination of the drinking water will have fallen below the permissible limit. Approximating the pore water in soil by distilled water and neglecting the possibility of formation of species of the heavy metals other than oxides and hydroxides is certainly a rough and preliminary procedure. It is promising for our work in this field that experimentally determined concentrations of heavy metals in equilibrium soil solutions are many orders of magnitude below the ones taken from Fig.4. A straightforward improvement of the simple procedure presented here is employing computerized chemical models, with which e.g. the simultaneous presence of many ligands can be considered. The deviation of the computer result from the experimentally determined one is primarily attributed to the unsufficient data base. In the last part of my paper (chapter IV) I discussed the immobilization of waste nuclides by adsorption on soil particles. Whereas molybdenum and technetium-appearing frequently as negatively charged species- are poorely sorbed by the negatively charged soil particles, many other heavy metals are readily sorbed by soil particles because of their positive charge. Now electrostatic adsorption is not the only process of immobilization. In Fig.7 potential locations of heavy metal sinks are summarized. The sinks that dominate the biologic availability of heavy metals are the iron/manganese-hydroxide coatings. And finally I mentioned the reverse process: the remobilization of the waste nuclides, when the coatings are re-dissolved as a consequence of a change of the chemical environment. We have then the effect of the decay of a "secondary repository" of the waste. The questions now are: How well isolated is that new repository from the biosphere,how quickly will it give off the waste nuclides?

The answers are still open to me. However I know of experiments that have shown that the remobilization can proceed in very short time, i.e. days. When we have all the necessary information on these processes, we can say, we need to contain e.g. nickel for say 10 half lives (10^5 years). And then we can ask the engineers, whether it is possible to build something that stays closed over that period. 10^5 years is too long, but if it would be 1000 years one might be optimistic. People working with concrete for hydropower dams are optimistic.

J.L.ANDERSON: What you say is when one considers all of these factors, that the radiowaste which would be generated in fusion plants must really be handled very carefully and that it may require deep burrial rather than shallow repository burrial because of the possibility of remobilization of some of these metals.

J.GRUBER: For drinking water the remobilized inventories might be less of a problem, since one expects the remobilized quantities to be small or the flow of water through the soil and consequently also the physical dilution to be large. But when the remobilization is induced by the roots of plants - they change their local environment to extract minerals from the soil- then the thermodynamically possible maximum concentrations might prevail closely around the roots and cause a considerable uptake of remobilized radionuclides.

J.L.ROUYER: Considering zirconium which appears below the ratio 1 of concentration, you assume that it is diluted in the environment and there is no need for a repository in this case?

J.GRUBER: Yes, but only if you term "dilution" the processes in environmental groundwater chemistry- as described in chapter III- that lead to the equilibrium concentrations c*.

J.L.ROUYER: But underlying this is a model of transfer and an assumption that the source term is diluted?

J.GRUBER: Yes. The graph (Fig.6) shows that one does not need physical dilution **for zirconium because it is not available** in a radiotoxic concentration in the soil solution. When one gives off active zirconium from a structure into the soil solution then the chemical processes reduce its concentration so quickly that one has a low concentration, lower than needed.

J.L.ROUYER: Because it is much below the ratio of 1?

J.GRUBER: Yes.

J.L.ANDERSON: But nickel and molybdenum do present potential problems?

J.GRUBER: Yes. My work during the last two years concentrated on nickel.

M.S.KAZIMI: Your explicit data refer to an entire structure made of nickel?

J.GRUBER: In Fig.1 I assumed 1 ppm of any element, i.e. also of nickel, in the structure. Fig.6 is by definition independent from the concentrations within the reactor structure since I used solubility data that have been calculated for the aqueous solution of one metal alone. The simultaneous presence of the other alloy constituents has been neglected.

M.S.KAZIMI: But it seems that if one would not use nickel or molybdenum in the first wall, the other things that you investigated would not cause harm?

J.GRUBER: Yes, this is true. If one avoids nickel or molybdenum containing structural material one is better off.

M.S.KAZIMI: What was the thickness of the first wall?

J.GRUBER: This is not relevant to my calculations: The amount of material that is activated does not play a major role here, because the concentration is the same in the soil solution regardless of the amount of corroded material. It is the amount of contaminated soil or water that depends on the amount of active waste. But in Fig.6 I discussed the concentration.

M.S.KAZIMI: I was thinking that if one would have more material that the concentration would be less.

J.GRUBER: The concentrations are fixed by environmental boundary conditions, not by the amount of corroding material.

M.S.KAZIMI: The concentrations in the soil?

J.GRUBER: In preequilibrium times one might get higher concentrations. But in the equilibrium situation, the soil cannot have higher concentrations because precipitation will occur. But the question on the kinetics is an important one.

T.K.ALLEN: Did you say you used ordinary 14 MeV neutrons?

J.GRUBER: I considered activation by fast neutrons and a thermal neutron spectrum for the two cases respectively. I did not consider, which is a drawback, the resonance reactions. I wanted to stay apart from details of the blanket design.

FUSION SAFETY AND PROTECTION ISSUES: LESSONS FROM FAST BREEDER REACTORS AND REPROCESSING PLANT STUDIES

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Abstract

The experience gained in the design and operation of nuclear reactors and fuel cycle facilities should be used to:

- develop a safety analysis which helps the designers and provides workers and public protection.
- provide means to respond to tritium problems
- study the non-radioactive risks.

This effort should give a high priority to technical evolution and innovations.

Introduction

As relative newcomers to fusion studies, we believe our best contribution to this meeting would be to apply to these studies our experience with fission reactors and fuel cycle facilities.

Our interest, like probably most of you, lies in wishing to develop a technology which - on a long term basis - will be able to meet the growing energy needs of mankind safely. Such a technology would help to assure good living conditions for the majority of the world's poor. Every means should be used for the objective. Here we think it of use to draw some pertinent lessons from the experience gained in the development of nuclear technology. This paper will deal with three topics

- comparison of design concepts
- tritium problems
- non-ionizing radiations and non-radioactive risks.

1. Comparison of design concepts

In reviewing the safety methodology used in the design phase of power reactors, it is necessary to remember that today's methods of safety assessment were not then available. The power reactor designs then were based on the requirement that the containment should withstand the pressure with an ample margin in hand. It was only in 1967 that Dr.Farmer of the U.K.A.E.A. presented a probabilistic approach to safety, and in the period 1972 - 74 that a team under Prof.Rasmussen carried out the first nuclear industrial application of this approach. Now this probabilistic total risk assessment is used whenever possible i.e. when data exist and the extent of the fault and event trees is not too great. The approach has built-in dynamic considerations about the rectification of failures. Despite the lack of modern safety assessment methods power reactors were designed.

In the case of fusion we are in both a better and a worse situation. A better, because there are still choices for the design. For worse, because we cannot apply a thorough probabilistic approach due to the lack of data and also the maximum accident approach is not satisfactory. Therefore there is a need for an approach that lies between the other two approaches.

Such a methodology applied to developing design concepts is possible and this Workshop should think about it. What should this method deal with?

Consideration of impact on the environment by the consequences of big accidents as for example current-arcs or lithium fires are certainly valuable (and information as presented by L.D. Muhlesstein regarding sodium fires give an advantage for fusion reactors, although lithium fires can last longer as there is more nitrogen than oxygen) although operating staff protection is omitted. This approach might be applicable to fission reactors assuming a time delay between the beginning of the accident and the point where operators come into danger although this is not true for all accidents. We cannot say that this approach applies to fusion mainly because of contamination by tritium.

In fact, WASH 1400 and I.I.A.S.A. studies on both fusion and fast breeder reactors are remarkable studies improving our knowledge of safety levels of these reactors but they are based on only one criterion - environmental safety.

Therefore, we have to visualize a safety-security analysis taking into account the protection of the public and of the workers. For worker protection the analysis should deal more with the operation accomplished (location, equipment used, normal and accident conditions encountered).

This analysis should allow for technological evolution and innovation. Design studies that we have done for reprocessing plants show that whenever one deals with a developing technology a constructive and anticipatory attitude is required. In the case of reprocessing plants we tried to get rid of the chemical plant engineer's approach of treating the new technological problems. We had to consider the replacement of large mechanical equipment in a radioactive environment not as a calamity but as a normal operation to be simply performed. This had implications on equipment and plant design and demanded new conditions for operators. For the equipment, we had to concentrate on maintenance problems due to mechanical failure, corrosion or circuit blockage. For the plant, we had to review the problems of equipment location and accessibility, redundancy, movement of waste, maintenance rooms, ventilation systems, etc.

We could not do as much as we wished because of the cost of some remote handling systems and the need to qualify such systems in a reasonable time. The analysis should also include some cost considerations. We have to assume that new technologies will develop but that this development will be weighted by some cost factor.

To summarize, we think that safety and protection methodology should be adapted to different design issues. Mathematical tools exist (multicriteria analysis used in pollution studies or theories related to uncertain environments) and groups in our Institute could contribute to such methods (A description of such a method is given in the Appendix to this paper). Basic data and some rough calculations about scenarios of credible accidents would be a starting point for comparing several credible design concepts.

2. Tritium problems

At this stage fusion reactors look like "hybrid" facilities from the safety and protection point of view. Under normal and accident conditions the failures of equipment characterizing reactor type conditions are often linked to the tritium contamination characterizing fuel cycle facilities. In the case of an arc-current, as described by P.Komarek, damage to neighbouring components would probably liberate tritium. Because of corrosion problems, maintenance operations - remote handled or not - will also create possibilities for tritium contamination somewhere in the plant.

Tritium problem is a challenge and we must develop new and improved safety tools.

Protection technology has limitations. For example, for the Joint European Torus (JET) where there is a purification and an isotopic separation system for exhaust gases coming from the machine, an emergency clean-up system for the tritium room has to be provided. This is possible if the amount of diluted gas to be treated is not too large.

JET is an experimental machine and therefore the problems to be dealt with are not completely applicable to fusion power reactors. They might be more difficult on experimental machines because the performances have to vary by a large amount (For fast breeders, RAPSODIE sodium is more contaminated than that of PHENIX) but thought must be given on concepts and the means to safely confine tritium releases.

JET experience gained during the design phase should be used after having separated out the problems specific to JET. The problems that K.Allen mentioned such as radiation damage to electronic components, monitoring and descommissioning aspects, and their proposed solutions should be of benefit to fusion development. In this field, non-classified data and information on technological improvements obtained with military facilities should be made available in civil applications. Data and codes for tritium transfer in the fast breeder reactors of CEA should be utilised.

3. Non-ionizing radiations and non-radioactive risks

Our Department of Protection has an important role in this field. Dr.Jammet, Head of that Department and President of the IRPA international committee on non-ionizing radiations (NIR) leads an international project relating to NIR and health protection problems. The dividing line between ionizing and non-ionizing radiation being a photon energy of 12 eV, corresponding to a wavelength of 100 nm. Their work deals with NIR matter interactions, NIR metrology, biological effects, exposure limits and guidelines for equipment performance. This Department also makes experiments notably on laser hazards related to eye exposure.

It is important - as Mr.J.G. Crocker said - to develop this non-radioactive aspect of safety and protection studies. Some of the risks might be found to be negligible, but certainly we will be able to show that some non-radioactive risks are very more dangerous causing causalities than are radioactive ones. Radioactive risks set new challenges, but comparison studies should continue.

Conclusion

The prospect for the development of fusion reactors appears good. Their development should benefit from past experience and we wish from the safety and protection point of view to co-operate with designers and operators of experimental fusion machines to contribute to improvements in this development.

APPENDIX

Identification of Research and Development needs in Safety Methodology.

A. Need for an overall safety analysis:

An overall safety method is needed now to accompany the development of fusion designs. The objectives would be:

- to provide an objective language between design engineers and safety specialists
- to eventually reject very bad (or promote very good) design concepts from the safety point of view
- to help determine where the money should go in Research and Development of safety programmes.

B. Definition of criteria:

The analysis should comprise multicriteria at the start. The criteria would be:

- number of manrem/year for workers.
- number of manrem/year for public.
- cost per KWh
- index of depletion of limited reserves (beryllium, etc.).
- index for waste disposal.

It is too early to unify such criteria now (for example, by providing designers with definite data on cost of a manrem or by rejecting some type of blanket material).

C. Definition of design concepts:

Design concepts to be compared should be determined by a method which make them realistic and comparable from the safety point of view. Realistic,means that efficient (same order of efficiency as in existing reactors or tritium plants) protective systems are included in the design (and such as the one suggested by H.Klippel are necessary). Comparable, means that there should not be too big discrepancy in design data produced for safety comparison.
A method such as the semiprobabilistic methodology described by F.Felicetti et al. (Safety Analysis. Semiprobabilistic Methodology and its Applicative Development; Report RT/DISP (78)10; Comitato Nazionale Energia Nucleare, Italy) should be envisaged, with the use of event trees, fault trees and minimal cut sets whenever possible and necessary. This method should be enriched by a sensitivity analysis of uncertainties. Criteria for judging the credibility of design concepts could be simpler than the above ones and more specific to radiation exposure.

D. Overall comparison and evolution of the method:

For the overall comparison, safety data will come from the phase of definition of design concepts. Comparison will be made with the above multicriteria. The method would be developing with improvements in accuracy of criteria and designs.

DISCUSSION

M.S.KAZIMI: Which is the methodology which has been applied to the safety analysis of SBX 1, the fast breeder reactor? As far as I know it relies mostly on the maximum possible accident, maximum credible accident concept. Which would be the lines you intend to follow for the second generation reactors, lets say SBX 2, which is at an early stage of design. This should provide a very good occasion, like the fusion reactor, to apply more deeply a probabilistic methodology.

J.L.ROUYER: As far as I know there were some more detailed studies regarding impact of accidents on the environment. The basic design concept is kept but improvements attempted with respect to some important components. It is also attempted to lower the costs of a following breeder reactor. But there is no major revision or review.

M.S.KAZIMI: You do not intend to move to a deeper use of probabilistic methodology?

J.L.ROUYER: For some parts of the plant only, e.g. iodine release. But there is not much of a problem.

M.S.KAZIMI: You pointed to the development of facility specific criteria. If I understood you correctly you were saying that we have to consider each facility as a special case and try to come out with particular criteria that are applicable in that case. Am I extrapolating what you said?

J.L.ROUYER: Not particularly for fusion reactors. In France there is two kinds of thinking. One is coming from the reactor people, WASH 1400, and one is coming from the fuel cycle facilities (reprocessing, fabrication plants) and this is actually a maximum credible accident line. For reprocessing we try to deepen the analysis and to built not such a simple methodology as maximum credible accident. To reactors on one hand and facilities on the other hand I relate two types of thinking.

J.G.CROCKER: I feel very strongly about the risk assessment. We worry about the ability to apply risk assessment in a licensing or safety analysis point of view. And sometimes we say, we cannot do it yet or the licensing people would not accept it or some agency will not accept this technology. And I think that this is, at this stage, the wrong approach to take. I think the real value in the risk assessment approach is for the individual system designer to understand the methodology. As a designer designs his system he can apply the risk assessment techniques to that particular system and then improve the design based on the results of his risk assessment. If he does that he will in fact cause one thing to happen that we often overlook, and that is the protection of the plant workers themselves. If you have designed a better system, and the designer has tested that system for failures, and if he says here are the weak links in that system based on his analysis and then he either adds another valve or adds whatever it takes to improve a particular design, he in fact has designed the safer system, which is going to help protect those people. That is something we now can do as we set out to design any system. We may not be able to force somebody or to accept a risk assessment methodology for the ultimate safety analysis but we can certainly design better machines based on that methodology.

J.L.ROUYER: I agree on your point: one must go the other way, as you said; one must design the system as not to fail, and so protect the worker, then attend to public protection .

J.G.CROCKER: I can give you a very good example of this technique. Both J.Anderson and I happen to be involved in this example right now. J.Anderson has a lot of experience in designing and operating tritium systems and has done that. However, in designing the TSTA, risk assessment approaches were not used. Most people do not. It is the very big exception rather than the rule. For the safety programme I am trying to push this methodology. So, here is one system that there is a complete design on, and we are going to do a risk assessment on it. So we are doing a full blown risk assessment on the TSTA and indeed what you will find from this risk assessment is that there are a number of places where you could make the system safer. But this is a wrong time to do it. Because one has to retrofit the system if one really wants to take advantage of it. We are not using it for that purpose but we are using it to illustrate the use of the technique itself.

J.L.ROUYER: But the design is made.

J.G.CROCKER: Yes. It is now too late to have any real impact.

J.C.CHICKEN: There is an additional point about risk assessment and the use of probability analysis that should be considered. When making such studies you should have in mind the reliability requirement the purchaser wants satisfied. The calculations you have to do to show that the reliability requirement is satisfied are exactly the same as you have to do to show the safety requirement is satisfied. The numbers are the same. If you have to sell the need for such analysis you have a powerful argument in the need for reliability analysis. The safety argument then is a spin-off from the reliability analysis.

J.G.CROCKER: I fully agree on that point.

HIGH PRIORITY FUSION SAFETY RESEARCH AND DEVELOPMENT NEEDS

I. TRITIUM HANDLING AND SAFETY

1. Characterization of realistic tritium releases during normal operation, maintenance and accident activities
2. Inventory of tritium in solid breeders
3. Tritium permeation rates - especially with regard to effects of implantation in the first wall and permeation barriers
4. Tritium conversion rate to oxide during routine and accidental releases
5. Efficiency of air clean-up systems following in-plant releases including releases accompanied by activation products
6. Adsorption and desorption rates on typical in-plant surfaces
7. Development of tritium trapping techniques that do not require conversion to oxide
8. Development of an electrolysis cell which is compatible with HTO and that can be used to recover tritium from aqueous waste streams

II. ACTIVATION PRODUCT GENERATION AND RELEASE

1. Assessment of realistic release mechanisms during normal operation, maintenance activities, and accidents
2. Development of low-activation materials for use in first wall and blankets
3. Development of believable maintenance scenarios for activated structures
4. Completed data base on the biological effects of activation product radionuclides

III. LITHIUM SAFETY

1. Establish data base for reactions of lithium compounds (Li_2O, Li_2SiO_3, $LiAlO_2$, lithium lead compounds, alloys, eutectics) with potential fusion reactor coolants
2. Establish correlation between large lithium spills and data base on small spills particularly with regard to lithium reaction mitigation and extinguishment
3. Develop and validate analysis models to determine effects of lithium reactions on fusion reactor structures
4. Examine corrosion effects on system components by lithium, and lithium compound aerosols
5. Develop effluent control systems for lithium or lithium compound aerosols containing activation products and/or tritium

IV. SUPERCONDUCTING MAGNET SAFETY

1. Provide experimental data on magnet response to off-normal/and accident conditions (quenches, arcs, shorting, lead failure, unequal currents)
2. Evaluate, and develop if necessary, the capability of existing computer codes to calculate magnet accident behaviour and the effects on the reactor components (this includes magnet quench behaviour, plasma disruptions and eddy current generation)
3. Develop safety systems to avoid and/or reduce the consequences of severe accident conditions
4. Plan for use of one or more of Large Coil Program coils in potentially severe safety tests at end of Large Coil Program
5. Provide experimental data on voltage/current characteristics of arcs under realistic conditions

V. POTENTIAL FUSION REACTOR ACCIDENT AND ACCIDENT ANALYSIS TECHNIQUES

1. Develop set of realisitic accident scenarios for fusion reactor systems
2. Develop comprehensive, time-dependent systems analysis codes for use in fusion reactor safety analysis
3. Develop fusion component reliability/failure data for use in fusion reactor risk assessment
4. Experiments should be done to quantify the important parameters (time scale, wall area, etc) of plasma disruptions
5. Development of probabilistic techniques as a method useable at any design stage (from concept to construction) for a systematic assessment of safety should be pursued

VI. OPERATIONAL SAFETY/SHIELDING CONSIDERATION

1. Develop set of accident scenarios that could lead to operator exposure to radiation and/or toxic materials
2. Evaluate adequacy of current one-dimensional, two-dimensional and three-dimensional shielding calculational techniques
3. Biological effects of magnetic fields do not appear to be a major problem; however, it is prudent that data be assembled or developed to support the establishment of defendable guidelines for magnetic field exposure limits
4. Evaluate adequacy of current operational radiation protection methods/practices for application in fusion reactors (e.g. neutron personnel monitoring; tritium monitoring instruments operating under a variety of environments)

VII. ENVIRONMENTAL IMPACTS

1. Data should be provided and analysis models developed for the following soil contamination interactions for activation products:
    - concentration in equilibrium soil solutions,
    - stability of chemical associations on soil,
    - dispersion, and
    - up-take into the food chain

VIII. WASTE MANAGEMENT

1. Develop techniques for disposal of routine releases of radioactivity from fusion plants, especially tritiated wastes
2. Develop realistic scenarios for storage of first wall and blanket waste materials
3. Develop strategy(ies) for disposal or recycle of fusion waste materials

Annex I

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Annex II

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