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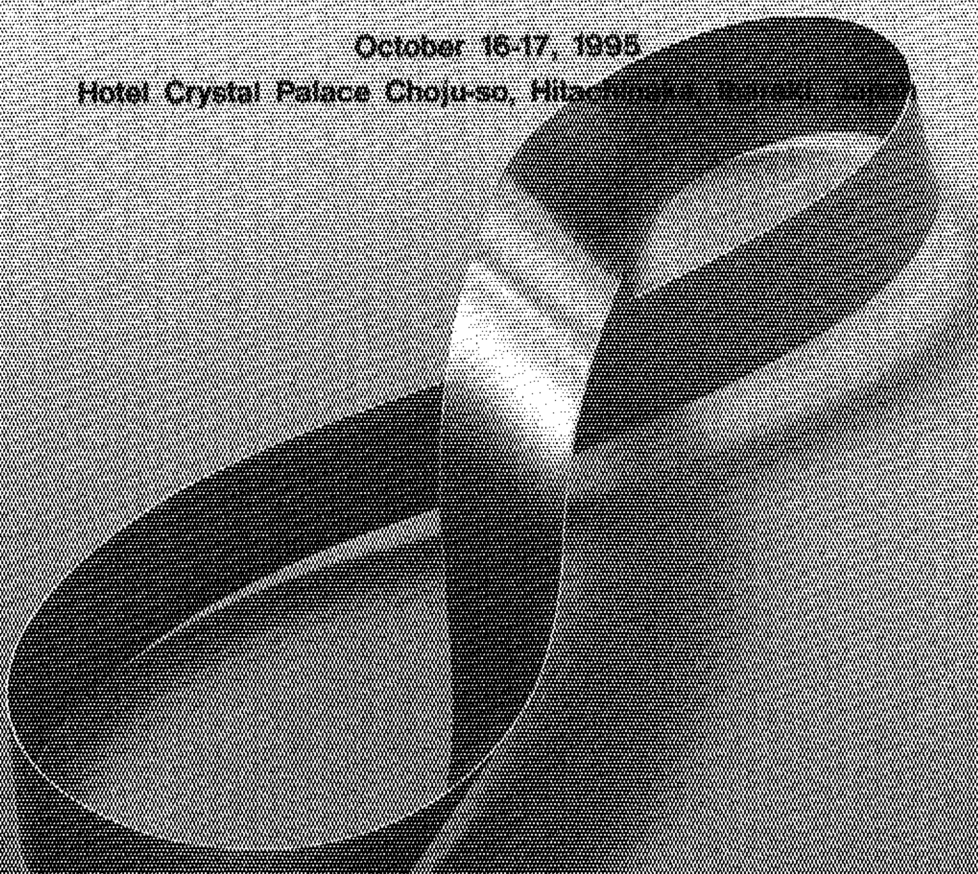
Proceedings of the First NUCEF International Symposium
NUCEF '95

—Engineering Safety of Nuclear Fuel Cycle Facility—

October 16-17, 1995
Hotel Crystal Palace Choju-so, Hitachinaka, Ibaraki, Japan



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Japan Atomic Energy Research Institute

in Cooperation with
SCIENCE AND TECHNOLOGY AGENCY
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
OECD NEA CSNI-WORKING GROUP ON FUEL CYCLE SAFETY

日 本 原 子 力 研 究 所
Japan Atomic Energy Research Institute

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NUCEF'95 Symposium Committee

Tokai Research Establishment
Japan Atomic Energy Research Institute
Tokai-mura, Naka-gun, Ibaraki-ken

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This volume contains all presented papers and the summary record of a panel discussion during the First NUCEF International Symposium (NUCEF'95) held October 16~17, 1995. The theme of the symposium was "Engineering Safety of Nuclear Fuel Cycle Facility". A total of 14 papers were presented in the six sessions: (1) Current Status and Important Issues on the Safety of Nuclear Fuel Cycle, (2) Prospects of Safety Research on Nuclear Fuel Cycle, (3) Outline of Safety Research of NUCEF, (4) Criticality Safety, (5) Accident Evaluation and Probabilistic Safety Analysis, and (6) Research on Reduction of Radioactivity Release and Database. The subject of the panel discussion was focused on the "Strategies for Safety Research in the Field of Nuclear Fuel Cycle and Effective Utilization of NUCEF".

Keywords: NUCEF, Nuclear Fuel Cycle, Safety Research, Criticality Safety, STACY, TRACY, Accident Evaluation, Fire and Explosion, Radioactivity Release

第1回NUCEF国際シンポジウム
NUCEF'95
―核燃料サイクル施設の工学安全―

1995年10月16～17日

茨城県ひたちなか市　ホテルクリスタルパレス長寿荘

報文集

日本原子力研究所東海研究所
NUCEF'95企画調整部会

（1996年1月24日受理）

本報文集は，1995年10月16～17日に開催された第1回NUCEF国際シンポジウム（NUCEF'95）における講演論文の全てとパネル討論の議事概要等を収録したものである。シンポジウムのテーマは「核燃料サイクル施設の工学安全」である。合計14件の論文が，(1)核燃料サイクルの安全性に関する現状と重要課題，(2)核燃料サイクルに関する安全性研究の展望，(3)NUCEFにおける安全性研究の概要，(4)臨界安全，(5)事故評価と確率論的安全解析，(6)放射能放出低減化に関する研究とデータベースの6セッションで発表された。また，パネル討論では，「核燃料サイクル分野における安全性研究課題とNUCEFの有効活用」が論じられた。

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Monday, October 16th; 1995

Opening Address

OPENING ADDRESS

Michio ICHIKAWA
Chairman of the NUCEF'95 organizing committee
(General Director of Tokai Establishment, JAERI)

Good morning. I am Michio Ichikawa of JAERI, chairman of the organizing committee. It is a great honor for me to welcome about 300 participants to this symposium.

The NUCEF started its hot operations at the beginning of this year. In this opportunity we have organized the First NUCEF International Symposium, NUCEF'95, entitled "Engineering Safety of Nuclear Fuel Cycle Facility".

We sincerely hope that all presentations and lively exchanges of information through this symposium will contribute to the progress of the research in the field of nuclear fuel cycle. I would like the delegates from various countries to feel at home here in Japan, and I hope their visit will be both pleasant and rewarding.

Now I would like to declare the start of this memorable symposium today and tomorrow.

Monday, October 16th, 1995

Welcoming Remarks

WELCOMING REMARKS
OF
THE FIRST NUCEF INTERNATIONAL SYMPOSIUM

Shozo SHIMOMURA
President, Japan Atomic Energy Research Institute

Good morning, ladies and gentlemen.

It is my pleasure to make welcoming remarks on the first NUCEF international symposium. First of all, I express my gratitude and extend my heartiest welcome to all the participants from home and abroad.

Japan Atomic Energy Research Institute has constructed NUCEF for safety research and fundamental research of nuclear fuel cycle, aiming at ensuring safety in the nuclear fuel recycling and innovating new fuel cycle technologies for the next century.

Construction of this research facility began six years ago, 1989, and completed in June 1994. Static Experiment Critical Facility (STACY) successfully achieved the first criticality in February 1995 and all the research works have been commenced.

As formulated in the "Long-Term Program for Research, Development and Utilization of Nuclear Energy" of our country, which was revised last year, 1994, the recycling of nuclear fuel is a basic element of our country's energy policy in its development and utilization of nuclear energy, from the viewpoints of ensuring the energy security, effective utilization of natural resources and preservation of the environment.

For that purpose, is being carried forward the construction of large scale reprocessing plant at Rokkasho Village in Aomori Prefecture, and are being studied the programs for recycling plutonium in an industrial scale.

Safety operation of nuclear fuel cycle facilities, including reprocessing plant, and appropriate management of radioactive wastes are important in the safe and effective recycling of nuclear fuel. This point of vital importance is clearly formulated in the above-mentioned Long-Term Program. Research and development for that purpose is one of the primary subjects to carry forward the recycling of nuclear fuel.

The role of NUCEF is to promote the safety research to support the safe and smooth commencement of Rokkasho reprocessing plant, and steadily to carry on the fundamental researches to enhance further the safety of nuclear fuel cycle including waste management by looking at the future.

Regarding the main topic of the symposium: “Engineering Safety of Nuclear Fuel Cycle Facility,” “criticality safety” research is one of the most important subjects from the viewpoint of facility design, operation and the concerns of public around fuel cycle facility. Effective utilization and contribution of NUCEF is vital for that purpose.

Not only “criticality safety” but also the generic safety issues of nuclear fuel cycle facility are great concern of public, home and abroad. I believe that this symposium will be of great significance from these points of view.

Together with ensuring safety, keeping transparency in research and development is another important aspect for carrying forward the recycling of nuclear fuel in our country. I expect that NUCEF will be a center of excellence which is opened and attractive to researchers and engineers in the field.

I wish that this symposium will afford an opportunity to promote the effective utilization of NUCEF and both domestic and international cooperative researches on the safety of nuclear fuel cycle facility.

I hope that all the participants will take the advantage of information exchange and discussions among the most active specialists from home and abroad, including the members of OECD/NEA Committee on the Safety of Nuclear Installations (CSNI)-Working Group on Fuel Cycle Safety.

As the closing remarks, I express my appreciation for the support of holding this symposium to the Science and Technology Agency, Power Reactor and Nuclear Fuel Development Corporation and OECD/NEA CSNI-Working Group on Fuel Cycle Safety, and also to the members of Organizing Committee who have played the leading role of this symposium.

THE FIRST NUCEF INTERNATIONAL SYMPOSIUM NUCEF'95 —WELCOMING REMARKS—

TAKAHIKO KONDO
Deputy Director General, Nuclear Safety Bureau,
Science and Technology Agency

Ladies and gentlemen, on behalf of our STA (Science and Technology Agency), the Government of Japan, I would like to briefly extend our sincere greetings to all of you here today, at this first NUCEF International Symposium NUCEF'95.

As we all know, development of nuclear power in the world has achieved a great progress in the past 50 years. It is, none the less, necessary for interested parties and specialists to make furthermore efforts hereafter so that nuclear power development shall be deployed steadily in its true sense. One of the most important aspects in the near future is the establishment of nuclear fuel cycle including reprocessing and waste management. As is cited in the "Long-Term Program for Development and Utilization of Nuclear Energy" revised last June by AEC (Atomic Energy Commission of Japan), promotion of research and development aimed at establishing the nuclear fuel cycle is deemed to be vital to attain the main goal.

For establishment of nuclear fuel cycle, it is naturally important to make efforts for people in each of different situations, such as industry people in safety designing, constructing and operation their facilities, regulatory people in strictly executing safety regulations and organization people in enhancing understanding of the public of the site as well as nationwide. However, it is the safety research that gives the ground and the support to all of these activities.

NUCEF was completed last June by the efforts of JAERI, and I have heard that researches have been started in full fledge, yielding significant results which are directed to improvement of safety, advancement of technology, enhancement of technical bases and others in the fields of reprocessing and waste management. I would sincerely expect that NUCEF be strenuously utilized by wide range of researchers home and abroad, not only those at JAERI but also from other various institutes.

In this context, I could say that it is very opportune to have the first NUCEF International Symposium held in cooperation with OECD NEA. Useless to say, the international cooperation as well as the domestic one utilizing NUCEF will take considerable time to obtain important results. In this symposium, I would expect that possibilities of cooperative works among national and international interests are sure to be discussed from various point of view, and I am confident that the first step towards the cooperation among home and abroad will be firmly started through these discussions.

In conclusion, I would like to say that NUCEF itself is only an instrument for performing researches, and whether it could be effective or not, depends solely on the will of you as a researcher. I wish this first NUCEF international symposium every success, and that the safety researches at NUCEF, further the safety researches for the nuclear fuel cycle will be proceeded successfully. Thank you.

WELCOMING REMARKS

KUNIHIKO UEMATSU
Director General of OECD/NEA

(Summary)

I have returned from OECD/NEA after serving for seven years as Director General of the Agency. During seven years, I have experienced many events, and I am sure, you have also experienced many events. Among those of events, I would like to point out just two this morning.

1. Slight return of confidence on the Nuclear Energy compared with the time of the Chernobyle accident. However, no full support of the nuclear energy returned to us. In the conclusion and re-commendation of the World Energy Congress which was held in Tokyo last week, it is strongly recommended that "government and industry should be undertaken to secure the public acceptability of nuclear power" and "we begin now".
2. Decline of the government interest in Nuclear Energy in some countries and also of the declination of their investment in nuclear research. This is not limited to nuclear energy field but it is also applied to the so-called Big Sciences. We are all experiencing the difficult time to obtain the government investment. Because of such tendency in the science field, we are experiencing the declined interest in science, especially in nuclear science, among young students.

What should we do now to solve these questions. I firmly believe that

1. We need to create new interest for youngs. This requires not only create new ideas but need to give new environment which they can work on. That means a facility is in need.
2. As you all aware that implementation of new idea and construction of a new facility are not an easy job at this time. I believe that the easiest and most sensible solution for this is to create a new international co-operative activity centered around an available new Facility.

I believe NUCEF is one of very important and rare new facilities available in the world for creating new international project. President of JAERI,

Mr.Shimomura and Mr.Kondo of STA have touched upon in this international cooperative idea this morning. I sincerely hope that this symposium would lead into a new international project. I shall give my full support for such an initiative.

Monday, October 16th, 1995

Plenary Session (1)

Current Status and Important Issues on the Safety of Nuclear Fuel Cycle

Cochairs
S.Matsuura(JAERI, Japan)
H.Auchere(IPSN, France)

DEVELOPING SAFETY IN THE NUCLEAR FUEL CYCLE

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Summary

The nuclear fuel cycle had its origins in the new technology developed in the 1940s and 50s involving novel physical and chemical processes.

At the front end of the cycle, mining, milling and fuel fabrication all underwent development, especially fuel fabrication but in general the focus of process development and safety concerns was the reprocessing stage, with radiation, contamination and criticality the chief hazards.

Here, the potential for major accidents is not as great as in nuclear fission reactors because of the lower available energy; but energetic events such as criticality, chemical explosions or overheating are possible and early safety studies focused on understanding and protecting against them.

In later decades the research and development studies have given us understanding of the mechanisms and physical events that could lead to accidents and developed a high level of confidence in the protection against major incidents. Issues such as accidental criticality, "red oil" explosion and high level waste tank overheating are examples.

Safety research is not over however and there is still work to be done in advancing technical knowledge to new generation nuclear fuels such as Mixed Oxide Fuel and in refining knowledge of margins and of potential upset conditions. Some comments are made on potential areas for work. The NUCEF facility will provide many useful data to aid safety analysis and accident prevention.

The routine operations in such plants, basically chemical factories, requires industrial safety and in addition the protection of workers against radiation or contamination. The engineering and management measures for this were novel and the early operation of such plants pioneering.

Later commissioning and operating experience has improved routine operating safety, leading to a new generation of factories with highly developed worker protection, engineering safeguards and safety management systems. Ventilation of contamination control zones, remote operation and maintenance, and advanced neutron shielding are engineering examples. In safety management, dose control practices, formally controlled operating procedures and safety cases, and audit processes are comparable with, or lead, best industry practice in other hazardous industries.

Nonetheless it is still important that the knowledge and experience from operating plants continue to be gathered together to provide a common basis for improvement.

The NEA Working Group on Fuel Cycle Safety provides a forum for much of this interchange. Some activities in the Group are described in particular the FINAS incident reporting system.

Introduction

The early development of reactor technology in the 1940s and 50s led to the establishment of first generation production reactors for generating electricity or for marine propulsion, the latter in a military context. Gas cooled and water cooled reactors were initially developed with a variety of fuels, coolants, and layouts. Over the years the prevalent thermal reactors have come to be water cooled reactors, the majority light water cooled, with uranium oxide fuel. Second generation gas cooled reactors are also oxide fuelled and so fuel cycle technology has focused on uranium extraction, fabrication and reprocessing with some plutonium fuel for fast reactors and plutonium recycle in mixed oxide fuelled thermal reactors. The development of this fuel cycle technology at industrial scale is what I will discuss.

Since this is the first NUCEF symposium I will try to mention safety research; I will also say something about the activities of the OECD/NEA Fuel Cycle Safety Working Group in promoting international exchange on fuel cycle safety. One can gain an interesting perspective on developments over the last fifteen years or so by comparing two state of the art reports on the safety of the nuclear fuel cycle produced by this group. One was published in 1981[1] and one in 1993[2].

Whilst glancing on all areas of the fuel cycle I will concentrate on reprocessing.

Uranium Mining and Milling

Originally the safety of mining and milling was seen as chiefly industrial safety. Some worker radiological protection was necessary, especially underground, and led to relatively high ventilation rates to sweep away radon in confined spaces. But with many ore bodies at concentrations below .1% of uranium human working is still practicable[3].

Recently significant ore bodies are being developed in Canada at uranium concentrations up to 15%. This is economically attractive, particularly in the current state of over supply of uranium, but human working in such mines is impossible. Remote mining methods will be necessary and radiation protection will become more stringent. It will be a worker safety issue not a public risk, and there are no major research needs.

The environmental impact of mill tailings is potentially very significant if they are not adequately retained[4]. Early experience included both leaks and leaching problems, and the subsequent history has been one of gradual improvement with more forethought put into retention. There is still a legacy in some areas of old tailings.

Conversion

The risks associated with conversion come from the use of corrosive and reactive agents such as hydrogen fluoride. This is not without its public safety aspects in the case of emergencies, but these are akin to emergencies at conventional chemical factories handling hazardous substances and there is wide experience of this.

Uranium Enrichment

The principal processes are gaseous diffusion and centrifuge enrichment. Laser enrichment is a recent possibility but not used on an industrial scale.

Both diffusion and centrifuge plant now operate continuously in effect, centrifuge technology is more economic and now has proven reliability of well over 99% per year without significant incident[5].

Enrichment of recycled uranium leads to slightly increased activity from uranium 234 and 232. Industrial scale operations have also shown some neptunium[6]. Technetium 99 may also be significant eventually. But given the low inventory, sealed, reliable plant there are not many important safety issues arising from this part of the cycle. The protection of enrichment tailings from external events involving fire can be an issue.

Fuel Fabrication

Oxide fuel production is well established at industrial scale in a number of countries (US, UK, France, Germany, Japan). Natural uranium is more a chemical than a radiological hazard, to combat this, the relatively primitive methods of dust control in the early plants have been improved on enormously in second generation facilities. Public risk is not significant.

Fast reactor fuel and mixed oxide fuel brings with it a range of more significant problems. Plutonium requires exacting safeguards. In addition there is significant neutron dose from plutonium 328 and 240 and gamma dose from ingrown Americium 241. Extremely secure containment of plutonium is essential to avoid inhalation. The methods developed are akin to those for plutonium handling at the back end of the fuel cycle and involve double containment, minimising glove box working and using remote operation where possible, and avoiding flammables, or build up of flammables such as hydrogen from radiolysis. The evolution of filtration technology and clean area philosophy has undoubtedly improved both the routine safety and accident resistance of plants.

In sintering furnaces hydrogen is used as a reducing agent and failure to control the furnace atmosphere could potentially lead to a hydrogen explosion. The technology is well understood however and has not resulted in incidents in practice; would be local only in any event.

Criticality probably remains the most serious potential accident, although it would affect workers rather than the public. With typical low enriched material criticality control is by mass and moderator control. More data at very low moderation could gain an increase in mass limits. For mixed oxide material the data on critical conditions are sparse. Research on critical quantities and dimensions for various MOX mixtures could enable margins better to be quantified.

The effects of a criticality incident at this stage are estimated in safety analyses, but there are a few data on moisture containing under moderated powders and research could improve knowledge of potential consequences.[7]

Spent Fuel Storage

There have been problems in the storage of metal fuel from gas cooled reactors, notably at Sellafield arising from a cessation of reprocessing and the consequent need to store for longer than anticipated. This led to particular problems with caesium in the pond water and a need to remove that to avoid excessive discharges. That is a metal fuel problem. For the bulk of oxide fuel, zircaloy or stainless clad, there are both wet and dry storage alternatives available. Both are used on a commercial scale. Some countries, notably Germany, have developed transport casks for dry storage.

Surveys of experience and understanding[8] have concluded that feasible and proven technologies exist already for both short and long term storage of spent fuel. Further research needs are probably limited to refinements of procedures such as re-racking in wet storage ponds. There may still be some room for refinement of source terms quantifying the potential releases following accidental draining of a wet storage pool. But the engineering of such facilities makes drainage an extremely low probability event associated with such things as earthquakes beyond the design basis. The need for source term refinement cannot therefore be considered to be great.

Gaining data on long term degradation mechanisms is part of the IAEA's BEFAST programmes.

Fuel Reprocessing

World wide reprocessing experience is already well over 10^5 thm of fuel; the vast bulk of this has been reprocessed by liquid/liquid extraction between nitric acid in water and TBP in kerosene, in the so called PUREX process.

Reprocessing is in many ways the natural area for concern in fuel cycle safety. Radioactive inventories are high, and the material is at most stages in a readily dispersible form. But a key difference from reactors, which also have a high inventory, is the dispersive energy available. In a reactor at power this is extremely large and can lead to consequences on a Chernobyl scale. In a reprocessing plant energies are much smaller and associated with local disruptive events such as criticality, fire, and explosion.

External events which may breach containment such as aircraft crash or earthquake also require some thought and engineering.

I will discuss briefly some of the hazards involved in this process and areas where safety research may have a part to play in improving safety or the efficiency of what is already an established industrial activity.

Criticality and Fissile Material

The nuclear fuel cycle is distinguished from other chemical processes handling toxic substances by two chief hazards: radiation and criticality. Radiation, a most important issue in worker protection is obviated by shielding. Well validated calculational methods are available.

Criticality is a complex phenomenon because of the interaction of the physical parameters such as absorption, fissile cross section, reflection. Not all are accurately known and calculation relies in practice on calibration from experimental measurements as close as possible to the system concerned. Simple calculational methods are based on handbooks that interpolate data points and are reliable and backed up by practical experience. But in industrial size plants quite complex computer codes are necessary to cope with large process quantities and all the operational circumstances. An example is that of uranium and plutonium nitrate solutions at relatively low enrichments. There are data on the relative components in solution but for mixed nitrate solutions process parameters established by calculation may still have over conservative safety margins and further measurements would support future MOX operations.

The problem areas are those where material location and quantity is unexpected or difficult to determine. An example could be the head shear pack under maintenance when moderator might mix with a quantity of retained fuel fines.

Many of the early criticality incidents (generally from the 1950s and early 60s) were caused by failures in the material control rather than gaps in the knowledge of critical systems. Typically mixing of undermoderated systems with moderator through stirring up solid residues in vessels or adding moderator. This has sometimes been associated (as in the Windscale incident in 1970) with an un-anticipated accumulation of material.

Whilst it is always the aim to avoid this by incorporating engineering safeguards it is normal practice to measure fissile inventory at key points. This is particularly difficult at the head end where the exact inventory of incoming fuel is subject to some uncertainty and the first accurate assay is in the accountancy tanks. In the dissolver it is hence common to use a neutron poison to ensure criticality safety. Poison is often expensive, such as gadolinium, and the amount added depends on the confidence one is prepared to put on knowing plant parameters such as the location of undissolved fuel, the burn up, reflection, and the nuclear cross sections.

It is an economic as much as a safety issue and since many of these plant parameters are very plant specific generic research is less likely to be helpful. Measurements of materials remaining in the hulls is normally by some technique such as neutron interrogation. Improved on-line assay methods here and in other process flows could improve operability and safety.

Another specific plant area where difficulties occur is in the waste area where material contaminated to a low level by transuranic waste must be taken and stored for later treatment. The assay of such low levels of fissile content is not particularly easy and leads to plant restrictions that are sometimes grossly conservative because one has to assume a fissile content at the level of resolution of the measuring equipment. (Similar issues apply in TRU waste treatment and disposal facilities). Development of measuring techniques is an area where research could bring improvements, as much in economics as safety.

Turning to the consequences of a criticality accident, these will in general consist of one or more pulses of radioactivity, perhaps accompanied by energetic effects and dispersion of material. Direct data are based on what can be deduced from early incidents together with experiments such as those conducted in the French CRAC facility. When using this information in safety analysis it is generally necessary to make assumptions that are conservative. In low enriched aqueous systems there may be scope to refine these by research and conservatism reduce. In heterogeneous systems where there is a possibility of a relatively rapid input of positive reactivity due to the criticality itself, it is not entirely clear that safety analysis assumptions are conservative.

Fires and Explosions

Kerosene is not a highly flammable solvent and most reprocessing plant operations are at relatively low temperature. It is not a problem provided there is good process protection against hydrocarbons passing with process flows into high temperature equipment such as furnaces. But there has been some concern, arising from both experimental work and past plant incidents, because it is possible to get rapid exothermic reactions from the oxidation of zircaloy fines or from the decomposition of mixtures of irradiated solvents, heavy metals, and nitric acid (sometimes called red oil explosions). The latter in particular has a complicated chemistry and there is not a complete fundamental understanding of the processes involved. However there has been a significant amount of research and development on both systems directed at establishing the boundary conditions within which rapid decomposition can occur. That enables plant equipment such as evaporators to be operated in a safe regime. It may be that further work would establish greater confidence and allow margins to be reduced with possible improvements in the efficiency of plant operation. However the complexities of the system are great.

External Events

External events such as wind, waves, aircrash and earthquake must be taken into account on an appropriate design basis, usually based upon the return period involved. This is the same approach as for reactors and other nuclear facilities and in general the predictive methods already developed can be used for plant siting. Engineering techniques already developed, particularly seismic engineering, can be carried into fuel cycle plant design to ensure that equipment failures caused by external events do not lead to release of material from the plant and that essential services such as coolant water are available at sufficiently high reliability. Some plant specific research or tests on equipment may be needed but further generic research is not.

Waste Issues

Some issues dealing with TRU have already been discussed. High level waste, or heat generating waste, has some special problems associated with it. The short term storage must have adequate heat rejection and is generally in liquid form. There is now relatively long term experience in this. Summaries drawing on the experience of the 70s and 80s[10] are still relevant. Most systems in use retain spare tank capacity to cope with an un-anticipated event such as corrosion failure, and also have highly redundant cooling systems to assure reliability. So whilst some years ago the boiling dry of a high level waste tank was considered an incident of potential concern, its probability is now recognised as being very low; in addition more recent studies[11] have shown that for beyond design basis accidents and boiling dry the bulk of activity release is delayed until the ultimate phase of approaching dryness when ruthenium in particular is volatilised. On the timescales involved emergency restoration of cooling by other means is almost certain and this accident sequence is hence of less concern.

Routine Emissions

Long lived isotopes, ^{85}Kr, ^{129}I, ^{14}C, were for some time an issue in reprocessing wastes. Krypton is not easily trapped, but recent studies on its radiological impact have shown it to be less important than originally thought[13], especially in view of the limited world reprocessing capacity. Established scrubbing and encapsulation techniques are available for iodine and carbon[14]. Whether long lived isotopes are trapped or not is generally considered to be an issue of what is practicable at reasonable cost. They do not present a high level risk level.

Encapsulation

Vitrification has now become an established technology with industrial scale plants based on French design operating in France and the United Kingdom[12]. The process involves much higher temperatures than other stages but no differences in principle. There are currently no operating repositories for high level waste but the principle of geological barriers is widely researched.

Encapsulation of medium level waste is still a live safety issue for long lived isotopes since their chemical form is related to final disposal needs. But waste issues are not the central subject of this symposium.

Learning from Experience

Safety research has over the years underpinned design and improved protection against accidents of any size, certainly against those accidents with the potential for affecting anyone off site.

In the case of worker risks however, much if not all the improvement in safety has come from experience with plants and refinement either in management methods or in engineering protection. As examples of management practice, the use of contamination control zones is now highly developed, the practice of dose budgeting for intervention and maintenance operations as well as routine operations have led to reduction in worker doses As engineering examples, additional shielding and second generation glove boxes or remote operation have had an impact on dose from dusty material and plutonium in particular.

In learning from experience it is obviously valuable to draw on as wide an experience base as possible. The Fuel Cycle Safety Working Group, acting within OECD/NEA as a sub group of the CSNI takes the exchange of experience as a regular agenda item. The aim is to ensure that all member countries are aware of significant evolutions in practice or of individual incidents or events that can be learned from. Not necessarily those of any significant consequence but those with significant potential consequence.

Over the last two or three years we have introduced the FINAS system in an attempt to systematise the flow of information and aid reporting and awareness easier. It is analogous in some ways to the IRS system for reactors. After a relatively slow start the system now has upwards of fifty events on it. We are currently reviewing its effectiveness and we hope to be able to make it an additional important mechanism for international exchange.

Routine operational safety is likely to show continuing improvement through the proper utilisation of experience both national and international. For example, corrosion and degradation processes are often revealed through experience rather than experiment because of differences between plant and laboratory conditions.

Advanced Reprocessing

A major change in the core technology away from the PUREX process would take a considerable time to evolve. This is not likely for the foreseeable future. But there are improvements that might be made, possibly in separation of TRU wastes or confinement of activity and trapping close to source. So the story is not yet over for safety research on process improvements.

Conclusion

The nuclear fuel cycle is established as a safe industrial process. Safety research will now bring incremental improvements and will quantify margins rather than leading to a step change. It may be valuable to do work in areas such as criticality of low enriched systems and mixed systems; assay, treatment, and handling of TRU material; and refinement of safety analysis methods dealing with hazards such as fires.

Exchange of operational experience internationally provides a broader basis for steady improvements in operational safety.

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Prospects of Safety Research on Nuclear Fuel Cycle

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SAFETY RESEARCH ON NUCLEAR FUEL FACILITY IN JAPAN

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SITUATION OF ENERGY RESOURCES IN JAPAN

Currently in Japan, the nuclear power is given a position to be the most crucial major energy in the near future, and the general system for its utilization must be righteously and properly deployed, according to the national basic policy for energy development.

In this context, efficient utilization of nuclear fuel resources is recognized to be, above all, vital for our country with scarce energy resources, so that the fuel cycle system with FBR (fast breeder reactor) should be hereafter established. Accordingly, in the current research and development on nuclear technology, attitude to head for realization of this fuel cycle system is firmly held nationwide. Figure 1 shows the overview of nuclear fuel cycle concept.

However, it is considered to take rather much time for the nuclear power utilization with FBR's to earn an economical justification, although depending on the forthcoming commercial situations of other energy resources. Until then, thermal reactors are continuously the main suppliers of nuclear power.

There may be one thought that recycling of nuclear fuel is not necessarily a requirement for nuclear power utilization with thermal reactors, and should be taken as one of possible options. However, to attain justification in the nuclear power utilization, complete treatment and disposal of all the materials generated from the use of nuclear power is considered to be an important problem, and reprocessing of spent fuel from thermal reactors is indispensable to solve the problem. Namely, only by reprocessing, one could find a safe method to finally dispose of the generated harmful material, and thus to obtain the public acceptance. In our country, the nuclear power utilization is basically proceeded in this line of consideration.

Nowadays, practically all of the world nuclear power totaling as much as 350 GWe, is generated by non-breeding thermal reactors, and its effective utilization ratio of nuclear fuel materials is as less than 1%. Moreover, because of the low utilization ratio, the amount of nuclear resources is limited to be as economically available as energy-cost balance sheets predict, so that the exhausting life time for the nuclear fuel resource is no more than that of the petroleum, provided the current technology applies. Such a situation as mentioned above, could not be drastically changed even if the recovered nuclear fuel material from reprocessing plants are to be used in LWR (light water reactor)'s. This is too an irrational way to use the resource in consideration of its enormous potentiality as energy resource.

On accomplishments of the FBR nuclear fuel cycle, such an irrationality could be directly rectified, and the resource utilization ratio would increase by two orders, as well as the amount of available nuclear resource would increase by more than two orders, corresponding to more than four orders increase of the exhausting life time, thus the possibility of nuclear fission energy as a practically end-less energy supplier come true for the benefit of the mankind.

From this point of view, in our country without any remarkable energy resources, the recovered nuclear fuel material from reprocessing spent fuel is situated to be as a semi-domestic energy resource, and is to be made of efficient use, and as a final destination, to be utilized in the FBR nuclear fuel cycle. And then, as the first step of D&D (Deployment and Development), reprocessing of the LWR spent fuels and utilization of the recovered fuel in LWR's are promoted as a link of the forthcoming recycle technology, aiming at its industrialization.

Current status of nuclear installations with their capacities are shown in Figure 2. In this figure, present and futuristic capacities (in parentheses) are illustrated for the uranium enrichment facilities, the uranium fuel manufacturing facilities, nuclear power stations, the spent fuel reprocessing facilities, the MOX (Mixed Oxide) fuel fabrication facilities and the radioactive waste treatment/disposal facilities.

DEPLOYMENT OF SAFETY RESEARCH IN JAPAN

Especially important are the LWR spent fuel reprocessing facilities, the MOX fuel fabrication facilities and the FBR spent fuel reprocessing facilities, as these facilities are all related to the nuclear fuel recycling usage. Accordingly, in the safety research in our country, the special emphases are placed on the improvement of reliability of safety assessments and on the reinforcement of safety assurance capabilities for these facilities.

Safety researches on nuclear installations have been undertaken to develop the safety assessment methods properly and without delay for the forthcoming nuclear technology, which will be definitely enlarged and diversified hereafter, although the safety of current facilities are adequate and basically secured, and further to improve rationality as much as possible by making safety margin quantified, and to realize ALARA (as low as reasonably achievable) attainability as highly as possible by improving safety assurance techniques and by developing advanced technologies.

These safety researches are designated to be planned, proposed, and executed mainly by government leadership, and are performed in national research organizations.

Such planning and proposition of safety researches are made by working out the five year annual plan, which is elaborately checked and reviewed by the specialists in each specialized research field under supervision of SCSRNI (Specialized Committee on Safety Researches of Nuclear Installations), which is suborganization of NSC (Nuclear Safety Commission),Japan. Namely, the fundamental rules for making the plan are worked out by SCSRNI, and then the working groups which are made up for each of the specially divided six research fields (water reactor, fast breeder reactor, nuclear fuel facility, radioactive material transport, seismic of nuclear installations, probabilistic safety analysis), as shown in Figure 3, are to review in detail the proposed research subjects from the national research organizations, and to evaluate them for the five year annual plan.

As for reviewing safety researches on nuclear fuel facilities, the working group has been formed, consisting of 19 members (4 from universities, 3 from JAERI (Japan Atomic Energy Research Institute), 5 from PNC (Power Reactor and Nuclear Fuel Development Corporation), 2 from other national research organizations and 5 from industries).

In addition to the safety researches, lots of safety demonstration tests are performed to certificate the soundness of safety security functions of important equipments/apparatuses of the facility by using the real scale mock-up test equipments. The planning, proposal and evaluation of these demonstration tests are made by the review committee consisting of members specialized in each area of subjects with support of the competent bureau of government related to the test objects.

Achievement of the safety assurance of nuclear fuel facility depends solely on how impeccable the safety design could be accomplished, and the safety design of facility is based principally on the multiple protection policy. Namely as shown in Figure 4, advocating the defense in depth concept, the followings are incorporated in the design of facilities, 1)prevention of the occurrence of the abnormal event during operation; 2)prevention of the expansion of the abnormal event and its development into an accident; 3)prevention of abnormal emission of radioactive material into peripheral environment. These technical measures are made certain by the administrative correspondence such as imposing more strict regulatory requirements together with specifying the safety related facilities and systems. On the other hand, the containment design is to be made by the multiple level/layer segregation principle to confine radioactive materials.

The safety design of nuclear fuel facilities as to correspondence with the peculiarity to nuclear events, is made to ensure the safety against criticality, shielding and confinement as impeccably as possible. For securing the confinement safety, it is vital to make the design to prevent fire and explosion, the seismic and shock-proof designs together with the facility design for radioactive waste treatment and disposal. Further, it is important to try to improve the safety more ardently from the viewpoint of safety operation, management, maintenance/repair, and from the radiation control point of view.

Safety research subjects for nuclear fuel facilities could be classified by the above mentioned viewpoints related to the safety assurance for facilities

FIVE YEAR ANNUAL PLAN FOR SAFETY RESEARCH

The nuclear fuel facility safety research five year annual plans were worked out four times in the past since 1976, and the current one has been valid from fiscal 1991 to 1995 as shown in Figure 5. And now, the next one, that is, the fifth five year annual plan starting at fiscal 1996 has been worked out as a result of about 1 year check and review. The work was initiated by deciding the "Basic Guidelines on Working-out of the Safety Research Annual Plan of Nuclear Installations" under discussion of the basic guideline committee. The nuclear fuel facility safety research annual plan has been worked out by check and review of the nuclear fuel facility working group according to the basic guidelines.

Basic Guidelines

In the basic guidelines as shown in Figure 6, firstly, the objective and scope for the safety research are clarified, next, the object facilities and its range of the research based on the annual plan are clearly described, and further, the procedure of how to work out the annual plan is indicated with requirements and important research area clarified.

Namely, the objective of the safety research is to "secure the safety of nuclear installations in preparation for the anticipated expansion and diversification of nuclear energy development and utilization, and to contribute to the formation of national consensus regarding the safety of the nuclear installations". Then, the scope of the safety research is to prepare the judgement material for use in the safety criteria, guides, and licensing safety review, and to improve the safety of the installations.

Additionally, the object installations for the research based on the annual plan are to be, "in principle, water reactors, fast breeder reactors, nuclear fuel facilities and radioactive transport casks". As for nuclear fuel facilities, reprocessing facilities, waste treatment/storage facilities, fuel fabrication/conversion facilities are cited as objects. And the scope of the researches covers the above facilities other than commercial, those difficult for the commercial to perform by the reason of tremendous load, and those needed to be executed from the long-range and wide-scope viewpoint.

Requirements for the annual plan to be worked out are to regard to promote positive international cooperations, to arrange to perform joint researches with universities as expected much concerning the fundamental researches, and to plot to make an effective communication with the safety demonstration tests using real scale equipments sponsored by the government from the viewpoint of complementary data exchange and synergistical use of the results.

And as research areas to be focused on, researches on human factors, severe accidents, probabilistic safety assessments, reprocessing, high aging effects and reliability of digital systems are cited in the basic guidelines.

Furthermore, the above mentioned six working groups, each working for water reactors, fast breeder reactors, nuclear fuel facilities, radioactive material transport, seismic response of nuclear installations and probabilistic safety assessment, are to do check & reviews for the committed areas as for working process concerning the annual plan. And hearings are to be made thoroughly from persons and organizations related to the proposed research subjects, thus preventing from missing necessary subjects by grasping the range of research generally and systematically. Finally, selection of research subjects to be described in the annual plan is to be done by attributing ranks to each of the proposed subjects in accordance to the judging criteria righteously set up from the importance and urgency point of view.

Notice that, once the annual plan is worked out, it is to be reviewed and revised as often as necessarily.

Procedures for Check and Review

According to the basic guidelines for the annual plan, the nuclear fuel facility working group has worked out the safety research annual plan for the nuclear fuel facilities after having made strenuous check and review starting from November 1994 ending in July 1995. Figure 7 shows how the anual plan have been worked out.

To begin with, the results of survey for research needs questioned among various research organizations and industries, related researchers and industry people, were checked and reviewed together with discussing the research subjects proposed by the national research organizations, being responsible for execution of the safety research described in the annual plan.

Namely, discussions were made after having had hearings of individual research subjects for each safety research area such as criticality safety, shielding safety, confinement safety etc. proposed by JAERI, PNC and other national responsible research organizations. On the other hand, quite a few ,more than 90, research subjects related to nuclear fuel facilities submitted as a result of the needs survey were discussed about their eligibility, feasibility, and necessity for the next annual plan. Further, as for 13 research subjects proposed by universities, the possibilities to accept as joint research programs for JAERI and PNC in addition to the suitability for the annual plan, were thoroughly checked and reviewed, and selected to be incorporated into the next annual plan, by referring to the results of combination of ranked evaluation for each of items such as objective suitability, crucialness, urgency and demands. Thus, the next annual plan has been worked out.

The detailed discussion of the contents of research subjects was made naturally by designating the technology development of nuclear fuel recycling and the steady progress of industrialization in our country. Namely, progress of construction of the Rokkasho reprocessing plant, incentive for industrialization of the MOX fuel fabrication, materialization of back-end measures such as for waste treatment and disposal, development of utilization program for high burn up fuel, progress of technology of reprocessing and others for establishment of the FBR nuclear fuel cycle, and instrumentation of research and development for innovative technology such as actinide recycle, these are all contributive to deployment of the nuclear fuel recycling program. Correspondingly, accumulation of basic data for use in preparation of judgement material for licensing safety review, and promotion of safety research to further improve the safety of facilities are becoming more and more important, and their research area is expanding.

In consideration of the situation of development in the nuclear fuel recycling technology, NUCEF of JAERI, which is just completed, would play a very crucial role, and expectation of deployment of the wide-ranged safety research on the safety assurance of nuclear fuel cycle related facilities, was pointed out to be most serious, especially on improvement of reliability of assuring criticality safety together with advancement of the recycling techniques including demonstration analyses of abnormal events and accidents.

On the other hand, PNC has accumulated data and experiences on design, construction, operation, maintenance and management of the Tokai Reprocessing Plant, the MOX fuel fabrication facility, the NINGYO-TOUGE uranium enrichment operation and the related radioactive waste treatment/storage facility. And the role of PNC by utilizing those data and experiences was also pointed out to be great in promoting researches contributing to advancement and rationalization of safety assessment data/methods applied to an innovative nuclear fuel recycling technology.

Current Research Subjects Adopted in the 5 Year Annual Plan

Safety researches on nuclear fuel facilities are, thus, to be performed for wide ranged research areas applying to diversified facilities. To select research subjects for the annual plan, the proposed research subjects were divided into five categories, namely, criticality safety, shielding safety, confinement safety, operation/radiation control with maintenance, and radioactive waste management, by arranging and synthesizing research elements included in the subjects.

Table 1 shows the selected research subjects. These are 4 from JAERI and 2 from PNC concerning criticality safety, 2 from JAERI, 1 from PNC and 1 from SRI (Ship Research Institute) concerning shielding safety, 4 from JAERI, 5 from PNC and 1 from NIMCR (National Institute of Materials and Chemical Research) concerning confinement safety, 2 from JAERI and 3 from PNC concerning operation control/maintenance and radiation management, 5 from JAERI and 6 from PNC concerning radioactive waste management. The last category was further subdivided into four subcategories, and the corresponding research subjects are 2 from JAERI and 1 from PNC concerning treatment of high level radioactive waste, 2 from PNC concerning treatment of low level radioactive waste, 2 from JAERI and 1 from PNC concerning treatment of TRU waste, finally 1 from JAERI and 2 from PNC concerning radioactive waste release performance and its reduction.

These subjects were selected in adequate consideration of the roles of each national research organizations having NUCEF for JAERI and the large scale testing facilities for PNC.

Furthermore, a lot of research subjects considered to be related to nuclear fuel facilities directly or indirectly, were selected in each of the working groups such as of radioactive material transport, of seismic study on nuclear installations and of probabilistic safety assessments. Several of the main subjects are shown in Table 2.

As for researches on radioactive waste, those related to its treatment and storage are to be handled in the nuclear fuel facility working group. Those related to its disposal are all handled by the other special committee on radioactive waste safety regulation, and correspondingly another safety research annual plan was worked out for researches on low level radioactive waste and high level radioactive waste. These contents are not cited here because of their volume.

Past and Continued Research Subjects Adopted in the 5 year Annual Plan

Quite a few of research subjects adopted in the annual plan for nuclear fuel facilities have been continuously performed for a relatively long duration. To illustrate the situation for references, Table 3 shows those adopted in the third annual plan valid from fiscal 1986 to fiscal 1990, and Table 4 shows those adopted in the fourth annual plan valid from fiscal 1991 and to fiscal 1995.

In the third annual plan, the working group was named as "the working group for nuclear fuel facilities such as reprocessing facilities" and fairly lots of research subjects were selected from wide range of aspects.

Namely, these were, firstly as research subjects common through all the nuclear fuel facilities, 1 from JAERI and 2 from PNC concerning criticality safety, 1 from PNC concerning shielding safety, 2 from JAERI and 1 from PNC concerning accident evaluation method, 3 from PNC concerning radiation management techniques and analysis/measurement techniques, 5 from PNC concerning operation management, maintenance/repair and inspection techniques. And next as related directly to reprocessing facility, were 1 from PNC & NRIM (National Research Institute for Metals) concerning corrosion safety, 5 from JAERI and 3 from PNC concerning safety of reprocessing process, 2 from JAERI and 2 from PNC concerning reduction of radioactive material release from reprocessing facility, 1 from JAERI and 1 from PNC concerning systematization of safety information for reprocessing facility, 2 from PNC concerning safety of plutonium handling facilities, 1 from JAERI and 4 from PNC concerning treatment and storage of high level radioactive wastes, 1 from JAERI and 3 from PNC concerning safety for treatment and storage of TRU wastes, 4 from PNC concerning safety of treatment and storage of radioactive wastes except high level wastes and TRU. And further, the probabilistic safety assessment working group selected 1 from JAERI and 3 from PNC as related to nuclear fuel facilities into the annual plan.

Next in the fourth annual plan as shown in Table 4, research subjects were a little bit more than those in the third annual plan, and were distributed to the eight areas. Firstly, they were 3 from JAERI and 2 from PNC concerning criticality safety, 1 from JAERI and 1 from PNC concerning shielding safety, 3 from PNC and 1 from NRIM concerning confinement safety, 3 from PNC concerning techniques of radiation management and analysis/measurement, 6 from PNC concerning operation management, maintenance/repair and inspection techniques, 5 from JAERI, 4 from PNC and 1 from NIMCR concerning accident assessment method, 3 from JAERI and 4 from PNC concerning reduction of radioactive material release, 3 from PNC and 1 from ONRI (Osaka National Research Institute) concerning high level radioactive waste treatment, 3 from PNC concerning low level radioactive waste treatment and 2 from PNC concerning TRU waste treatment. And further, the probabilistic safety assessment working group selected 1 from JAERI and 3 from PNC as related to nuclear fuel facilities into the annual plan.

CONCLUDING REMARKS

Summarizing the safety research subjects adopted in the past two and the current annual plan, one can see lots of them are the same or similar and have been continuously selected, indicating that the national safety researches have been performed persistently in accordance with the progress of technology. Figure 8 shows examples of relation of each theme of safety researches with nuclear fuel cycle facilities.

Regarding the above mentioned safety demonstration tests, the safety demonstration tests on the reprocessing plant and others have been performed under the auspices of the Science and Technology Agency as shown in Table 5. Out of 13 test subjects in this Table, 8 have been already finished and 5 are now under execution. These demonstration tests have been aimed at demonstration of the safety of reprocessing technology by using real scale mock-up test facilities for PA (Public Acceptance), and the engineering data obtained from these tests have been very useful as judgement material for licensing safety review of the applied facility.

Further in the case of difficulty to demonstrate safety by experiments such as for credible accidental events, computer codes for use in demonstration of the safety have been prepared at JINS (Institute of Nuclear Safety, Japan) and NUSTEC (Nuclear Safety Technology Center).

On the other hand, universities are conducting a lot of fundamental science and engineering researches on nuclear safety as a voluntary base, and moreover participating in individual safety research annual programs hosted by JAERI or PNC as a joint research base.

Furthermore, commercial industries with nuclear fuel facilities and their supporting contractors are conducting many of the voluntary safety assurance tests for implemented technology to secure safety of the facilities.

As mentioned above, nuclear fuel facilities in our country, having experienced more or less troubles, have not suffered from a serious accident so far, which might have influenced the public health, thus could be said to have secured their safety. Hereafter, in the anticipated further progress of nuclear facilities in our country, which will be characteristic of full scale utilization of recycled plutonium, I firmly believe that the safety of the nuclear facility will be secured undoubtedly if the ardent attitude shown so far in pursuit of securing safety including safety research, on concluding my presentation entitled as "Safety Research on Nuclear Fuel Facility in Japan".

Table 1−(1/2) Safety Research Subjects in the 5th Annual Plan for Nuclear Fuel Facilities(1996 ～ 2000) (1)

1. Research on Criticality Safety	
(1) Research on Criticality Safety Evaluation Method	(JAERI)
(2) Experimental Research on Criticality Safety	(JAERI)
(3) Research on Process Criticality Safety	(JAERI)
(4) Development of Subcriticality Measurement System	(PNC)
(5) Evaluation of Nuclide Composition for Spent Fuel	(JAERI)
(6) Research on Criticality Control for MOX Fabrication Facility and Others	(PNC)
2. Research on Shielding Safety	
(7) Preparation of Nuclear Data Needed for Radiation Source Evaluation and Others	(JAERI)
(8) Construction of Criticality / Shielding Numerical Experiment System	(JAERI)
(9) Research on Neutron Dose Evaluation for Nuclear Fuel Facilities	(PNC)
(10) Experimental Research on Safety Margin Evaluated by Shielding Design Method for Complex Geometry	(SRI)
3. Research on Confinement Safety	
(11) Research on Aerosol Source Term Evaluation	(JAERI)
(12) Research on Phenomena of Abnormal Chemical Reaction in Nuclear Fuel Cycle Facilities	(JAERI)
(13) Research on Evaluation Method for Reprocessing Process Abnormality	(JAERI)
(14) Test and Research for Abnormal Phenomenon Assessment	(PNC)
(15) Development of Structural Safety Analysis Program for Nuclear Fuel Facilities	(JAERI)
(16) Research on Explosion Source for Anti−shock Assessment for Nuclear Installations	(NIMCR)
(17) Safety Test of Glove Boxes and Others	(PNC)
(18) Research on Safety Handling Techniques for MOX Fuel Powder	(PNC)
(19) Safety Research on Hydrogen Mixed Gas	(PNC)
(20) Study on Equipment Having Passive Safety Function Applied for Nuclear Fuel Facility	(PNC)

Table 1−(2/2) Safety Research Subjects in the 5th Annual Plan for Nuclear Fuel Facilities(1996 ～ 2000) (2)

4. Research on Operation Control / Maintenance and Radiation Management	
(21) Resarch on Improvement of In−service Inspection Technology	(PNC)
(22) Reseach on Anti−radiation Property of Electric / Electronic Parts	(PNC)
(23) Development of Monitoring System for Abnormal Events of Reprocessing Facilities	(JAERI)
(24) Development of Monitoring Techniques for Radioactive Materials	(JAERI)
(25) Research on Sytem Development of Radiation Monitoring and Management in Reprocessing Facilities	(PNC)
5. Research on Radioactive Waste Management	
1) Research on Treatment of High Level Radioactive Waste	
(26) Research on enhanced Volume Reduction of High Level Waste	(PNC)
(27) Basic Tests on Four Group Partitioning Process Safety	(JAERI)
(28) Safety Research on Advanced Reprocessing Process	(JAERI)
2) Research on Treatment of Low Level Radioactive Waste	
(29) Research on Waste Materialization Including Iodine	(PNC)
(30) Research on enhanced Volume Reduction Technique for Various Low Level Radioactive Liquid Waste	(PNC)
3) Research on Treatment of TRU Waste	
(31) Research on Non−destructive Measurement Technique for TRU Waste	(PNC)
(32) Research on Advanced Treatment Technique for TRU Waste	(JAERI)
(33) Research on a New Solidification Treatment Technique for TRU Waste	(JAERI)
4) Research on Release Behavior Characteristics and Reduction of Radioactive Waste	
(34) Research on Gaseous Phase Transfer and Treatment of Volatile Nuclides	(JAERI)
(35) Development of Advanced Technologies for Removing Iodine	(PNC)
(36) Research on Krypton Recovery and Solidification	(PNC)

Table 2 Safety Research Subjects Related to Nuclear Fuel Facilities in the 5th Annual Plans for Other Fields

1. Radioactive Material Transport
   (1) Research on Criticality Safety of Spent Fuel Transport Package Considering Burn−up (JAERI)
   (2) Research on Neutron Shielding of High Burn−up Spent Fuel Transport (SRI)

2. Seismic Aspects of Nuclear Installations
   (1) Research on Seismic Safety Margin Evaluation Method for Aging of Equipment and Piping System (DPRI)
   (2) Research on Seismic Isolators (JAERI)

3. Probabilistic Safety Assessment
   (1) Research on PSA Methods for Reprocessing Facilities (JAERI)
   (2) Research on Reliability Assessment Methods for Nuclear Fuel Facilities (PNC)
   (3) Application Study of Probabilistic Safety Assessment for Nuclear Fuel Facilities (PNC)

Note) (JAERI): Japan Atomic Energy Research Institute
(PNC): Power Reactor and Nuclear Fuel Development Corporation
(SRI): Ship Research Institute
(NIMCR): National Institute of Materials and Chemical Research
(DPRI) : Disaster Prevention Research Institute

Table 3−(1/3) Safety Research Subjects in the 3rd Annual Plan for Reprocessing Plants and Other Nuclear Fuel Facilities(1986 ~ 1990) (1)

[1] Research Areas Common to Nuclear Fuel Facilities	
1. Research on Criticality Safety	
(1) Criticality Safety Experiment	(JAERI)
(2) Japan−U.S. Joint Critical Experiment and Development of Subcriticality Measurement Techniques	(PNC)
(3) Research on Criticality Control of Plutonium Handling Facilities	(PNC)
2. Research on Shielding Safety	
(4) Research on Nuetron Shielding and Radiation Exposure Management	(PNC)
3. Research on Accident Evaluation Method	
(5) Research on Source Term Evaluation for Radio−activity Release at an Accident	(JAERI)
(6) Development of Accident Analysis Code for Fuel Reprocessing Plants	(JAERI)
(7) Research on Safety of Large Cells	(PNC)
4. Research on Radiation Management Techniques and Analysis / Measurement Techniques	
(8) Development of Automatic and Remote Measurement Techniques for Radiation Management	(PNC)
(9) Development of In−line Measurement Techniques	(PNC)
(10) Development of Sensors and Radiation Resistant Materials	(PNC)
5. Research on Operation Management, Maintenance / Repair and Inspection Techniques	
(11) Research on Reliability of Remote Operation Techniques	(PNC)
(12) Development of Preventive Maintenance System for Facilities	(PNC)
(13) Development of In−service Inspection Techniques	(PNC)
(14) Research on Advanced Remote Maintenance Tachniques Using Rack System	(PNC)
(15) Development of Instruction System for Reprocessing Facility Operation	(PNC)

Table 3-(2/3) Safety Research Subjects in the 3rd Annual Plan for Reprocessing Plants and Other Nuclear Fuel Facilities (1986 ~ 1990) (2)

[2] Research Area for Reprocessing Facilities	
6. Research on Corrosion Resistance Safety in Reprocessing Facilities	
(16) Research on Corrosion Resistance Safety of Structural Material and its Welding method	(NRIM & PNC)
7. Research on Safety of Reprocessing Process	
(17) Research on Evaluation of Aerosol Properties and Collection Technigues at Fuel Assembly Dismantling by Laser Beam	(PNC)
(18) Safety Research on Dissolution of Spent Fuel	(JAERI & PNC)
(19) Safety Evaluation test on Abnormal Operation Behavior of Extraction Process	(JAERI & PNC)
(20) Research on Radiation Damage on Reprocessing Extraction Solvents	(JAERI)
(21) Research on Hydrogen Generation in Reprocessing Process	(JAERI)
(22) Development of Simulation Program for Reprocessing Process	(JAERI)
8. Research on Reduction of Radioactive Material Release from Reprocessing Facility	
(23) Development of Radioactive Iodine Removal and Storage Techniques	(JAERI & PNC)
(24) Development of krypton Recovery and Storage Techniques	(PNC)
(25) Development of Tritium Release Reduction Techniques	(JAERI)
9. Research on systematization of Safety Information for Reprocessing Facility	
(26) Development of Safety Data Base for Reprocessing Facility	(PNC)
(27) Preparation of Safety Evaluation Handbook for Reprocessing Facility	(JAERI)
[3] Research Areas for Plutonium Handling Facilities	
10. Research on Safety of Plutonium Handling Facilities	
(28) Research on Containability of Plutonium	(PNC)
(29) Test for Comprehensive Safety Demonstration of Plutonium Handling Facilities	(PNC)

Table 3−(3/3) Safety Research Subjects in the 3rd Annual Plan for Reprocessing Plants and Other Nuclear Fuel Facilities (1986 ～ 1990) (3)

[4] Research Area for Treatment and Storage of Radioactive Wastes	
11. Research on Treatment and Storage of High Level Radioactive Wastes	
(30) Research on Reliability of Liquid Waste Pretreatment Techniques	(PNC)
(31) Research on Safety Measures for a Glass Melting Furnace	(PNC)
(32) Research on Improvement of Removal Performance of Radioactive materials in Off−gas	(PNC)
(33) Research on Safety of Storage Facilities	(JAERI & PNC)
12. Research on Safety for Treatment and Storage of TRU Wastes	
(34) Research on Reliability of Reduction Techniques for Generation of TRU Wastes	(PNC)
(35) Research on Safety and Reliability of Volume Reduction and Storage Techniques for TRU Wastes	(PNC)
(36) Research on Safety of TRU Waste Management	(PNC)
(37) Research on Safety Evaluation of TRU Waste Treatment and Storage	(JAERI)
13. Research on Safety for Treatment and Storage of Radioactive Wastes Except High Level Wastes and TRU	
(38) Research on Reliability of Volume Reduction Techniques for Radioactive Waste	(PNC)
(39) Research on Reliability of Incineration Techniques for Noncombustible Radioactive Waste	(PNC)
(40) Research on Reliability of treatment Techniques for Analysis Waste and Salt−containing Waste Solution	(PNC)
(41) Research on Integrity of Asphalt Solid and Plastic Solid	(PNC)
[Related Safety Research Subjects in the Annual Plan for Probabilistic Safety Assessments]	
(1) Improvement and Development of Reliability Evaluation Methods for Reprocessing Facilities	(PNC)
(2) Development of Accident Analysis Method for Reprocessing Facilities	(JAERI)
(3) Execution of Probabilistic Safety Assessment for Reprocessing Facilities	(PNC)
(4) Application of Probabilistic Safety Assessment to Nuclear Fuel Facility Technical Criteria	(PNC)

Table 4-(1/3) Safety Research Subjects in the 4th Annual Plan for Nuclear Fuel Facilities (1991 ~ 1995) (1)

1. Research on Criticality Safety	
(1) Development of Subcriticality Measurement System	(PNC)
(2) Research on Criticality Safety Evaluation Method	(JAERI)
(3) Preparation of Criticality Safety Data	(JAERI)
(4) Research on Process Criticality Safety	(JAERI)
(5) Research on Criticality Control for Plutonium Handling Facility	(PNC)
2. Research on Shielding Safety	
(6) Research on Neutron Dose Evaluation for Nuclear Fuel Facilities	(PNC)
(7) Research on Shielding Safety Evaluation Methods	(JAERI)
3. Research on Confinement Safety	
(8) Research on Corrosion Safety of Structural Material and its Fabrication Method	(PNC)
(9) Research on Corrosion Integrity of Diffenent Material Juncture for Reprocessing Facilities	(NRIM)
(10) Research on Plutonium Confinement Functions	(PNC)
(11) Research on Fluoride Utilization and Long Term Safe Storage of Depleted Uranium	(PNC)
4. Research on Techniques of Radiation Management and Analysis/Measurement	
(12) Research on Sensors and Para-Radiation Resistance Materials	(PNC)
(13) Research on Optimization of Radiation Protection for Nuclear Fuel Facilities	(PNC)
(14) Research on Continuous Monitoring Techniques for Radioactive Material in Exhaust-gas	(PNC)

Note) (NRIM) : National Research Institute for Metals

Table 4−(2/3) Safety Research Subjects in the 4th Annual Plan for Nuclear Fuel Facilities (1991 ～ 1995) (2)

5. Research on Operation Management, Maintenance/Repair and Inspection Techniques.	
(15) Research on Improvement of Reliability for Remote Operation Techniques	(PNC)
(16) Research on Improvement of In−service Inspection Techniques	(PNC)
(17) Research on Improvement of Reliability for Remote Piping Joint	(PNC)
(18) Research on Diagnostie Techniques for Abnormality of Extraction Process	(PNC)
(19) Research on Handling of Unresolved Residues	(PNC)
(20) Basic Research on Depletion of Solvent	(PNC)
6. Research on Accident Assessment Method	
(21) Evaluation of source Term of Criticality Accidents	(JAERI)
(22) Development of Safety Assessment Method for Accidents	(PNC)
(23) Research on Assessment of Process Characteristics at Abnormal Transient	(JAERI)
(24) Research on Aerosol Source Term Evaluation	(JAERI)
(25) Research on Evaluation Techniques for Behavior of Air−born Radioactive Materials	(PNC)
(26) Research on Hydrogen Generation Behavior of Solution in Storage Vessel of Reprocessing Facilities	(PNC)
(27) Research on Radiation Damage on Reprocessing Extraction Solvent	(JAERI)
(28) Research on Rapid Burning by Nitro Reaction of Organic Material	(AIST)
(29) Research on Plutonium Source Term during Accidents of Reprocessing Facilities	(JAERI)
(30) Research on Handling Safety of Metallic Plutonium	(PNC)
7. Research on Reduction of Radioactive Material Release	
(31) Research on dissolution of Spent Fuel	(PNC)
(32) Research on dissolution of High Burned−up Fuel	(JAERI)
(33) Research on Adsorbers of Radioactive Iodine	(PNC)
(34) Research on Removal / Storage Techniques of Radioactive Iodine	(JAERI)
(35) Research on Recovery / Storage Techniques of Krypton	(PNC)
(36) Research on Reduction of Radioactive Liquid Release by Closed System	(JAERI)
(37) Research on Handling Safety of Recovered Uranium	(PNC)

Table 4−(3/3) Safety Research Subjects in the 4th Annual Plan for Nuclear Fuel Facilities (1991 ～ 1995) (3)

8. Research on Radioactive Waste Treatment
 1) Research on High Lebel Raioactive Waste Treatment
  (38) Research on Material Property Assessment for Vitrified Solid (PNC)
  (39) Research on Inspection Techniques for Vitrification Melting Furnace (PNC)
  (40) Research on Partitioning Phenomena during Glass Melting (ONRI)
  (41) Research on Reliabilities of Volume Reduction Techniques for Radioactive Waste (PNC)
 2) Research on Low Level Radioactive Waste Treatment
  (42) Research on Safety of Solidified Material of Incombastible Solid Waste (PNC)
  (43) Research on Separation Techniques for Radioactive Nuclides from Low Level Radioactive Concentrated Waste Solution/Solvent (PNC)
  (44) Development of Handling/Treatment Techniques of Radioactive Waste Generated from Recovered Uranium Conversion (PNC)
 3) Research on TRU Waste Treatment
  (45) Research on Safety of Volume Reduction Treatment/Storage for TRU Waste (PNC)
  (46) Research on Safety of TRU Waste Management (PNC)

[Related Research Subjects in the Annual Plan for Probabilistic Safety Assessment]

  (1) Development and Improvement of Safety Evaluation Methods for Reprocessing Facilities (PNC)
  (2) Research on Sounce Term Evaluation Methods for Reprocessing Facilities (JAERI)
  (3) Application Study on Probabilistic Safety Assessment for Reprocessing Facilities (PNC)

Note) (ONRL) : Osaka National Research Institute

Table 5 Safety Demonstration Tests and Demonstration Analyses for Reprocessing and Other Facilities

Demonstration Tests

(1) Demonstration Tests for Corrosion Safety of Reprocessing Facilities
(1980 ～ 1987 ; Sumitomo Chemical Co. LTD.)

(2) Demonstration Tests for Exhaust Filters Safety at a Severe Condition for Reprocessing Facilities
(1981 ～ 1986 ; Daido Steel Co. LTD.)

(3) Demonstration Teats for Exhaust Filters Safety at an Accident Condition for Reprocessing Facilities
(1981 ～ 1985 ; JAERI)

(4) Demonstration Tests for Criticality Safety at Reprocessing Facilities
(1985 ～ (1999) ; JAERI)

(5) Demonstration Tests for Seismic Safety for Glove Boxes
(1986 ～ 1988 ; Mitsi Engineering & Shipbuilding Co. LTD.)

(6) Safety Demonstration Tests for Cell Ventilation System at Reprocessing Facilities
(1986 ～ 1997 ; JAERI)

(7) Safety Demonstration Tests for Off−gas Adsorption System of Reprocessing Facilities
(1987 ～ 1991 ; NUSTEC)

(8) Demonstration Tests for Confinement Safety of Vitrified Solid Waste

(9) Demonstration Tests for Corrosion Safety of New Materials of Reprocessing Facilities
(1988 ～ 1994 ; Sumitomo Chemical Co. LTD.)

(10) Demonstration Tests for Extraction Process Safety of Reprocessing Facilities (1989 ～ 1992 ; JAERI)

(11) Demonstration Tests for Containment Process Safety for Radioactive Nuclides (1993 ～ (1997) ; JAERI)

(12) Demonstration Tests for Safety Margin of Spent Fuel Nuclide Composition (1994 ～ (1998) ; JAERI)

(13) Demonstration Tests for Corrosion Safety of Advanced Matenials in Reprocessing Facilities (1995 ～ (2004) ; JAERI)

Demonstration Analyses

(1) Preparation of Safety Analysis Program for Nuclear Fuel Facilities (1980 ～ ; JINS)

(2) Demonstration Analysis for Safety of Reprocessing and Other Facilities (1986 ～ ; JINS)

(3) Accident Evaluation for Safety Demonstration of Reprocessing Facilities (1987 ～ 1991 ; NUSTEC)

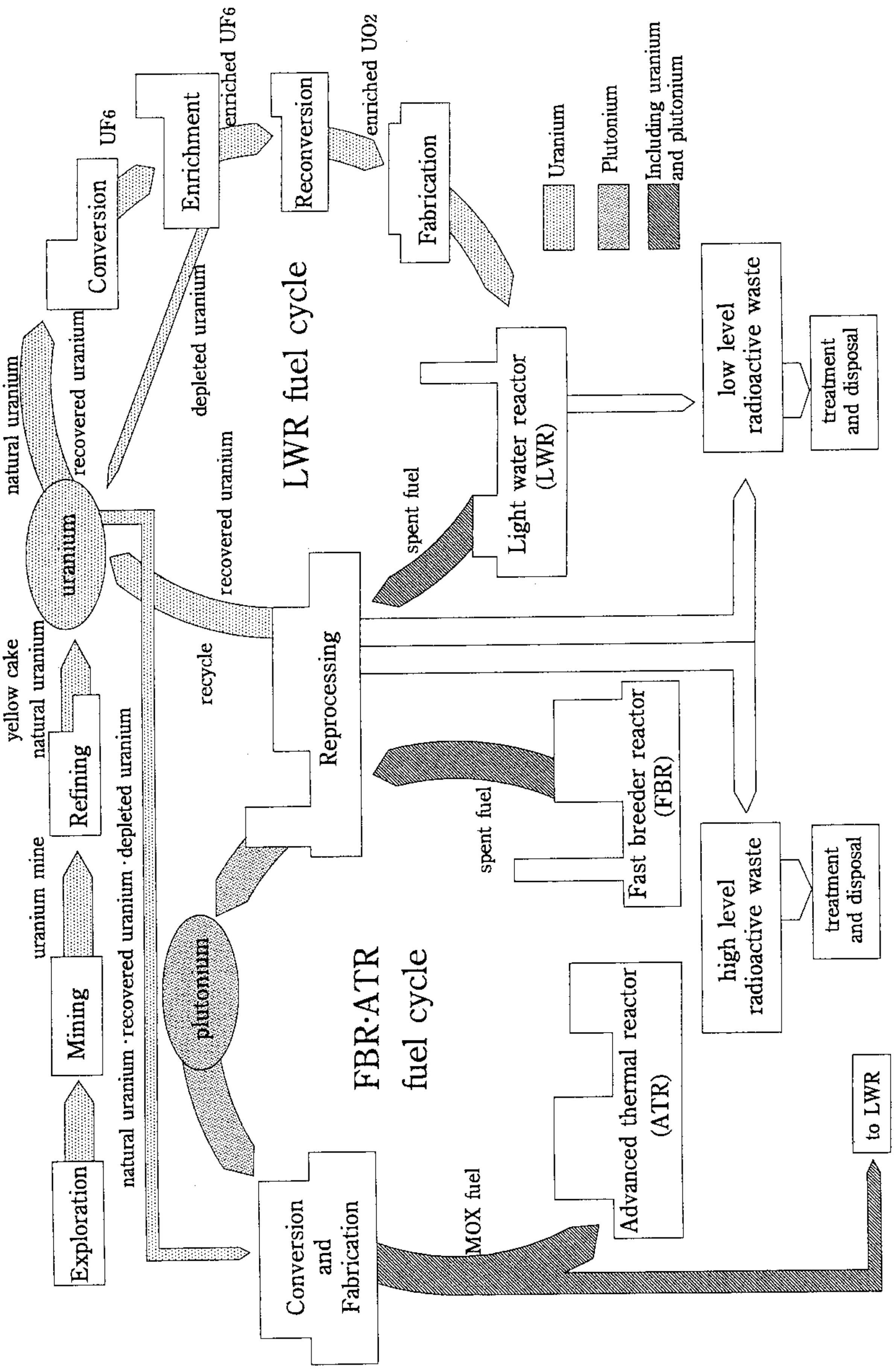

Figure 1 Overview of Nuclear Fuel Cycle

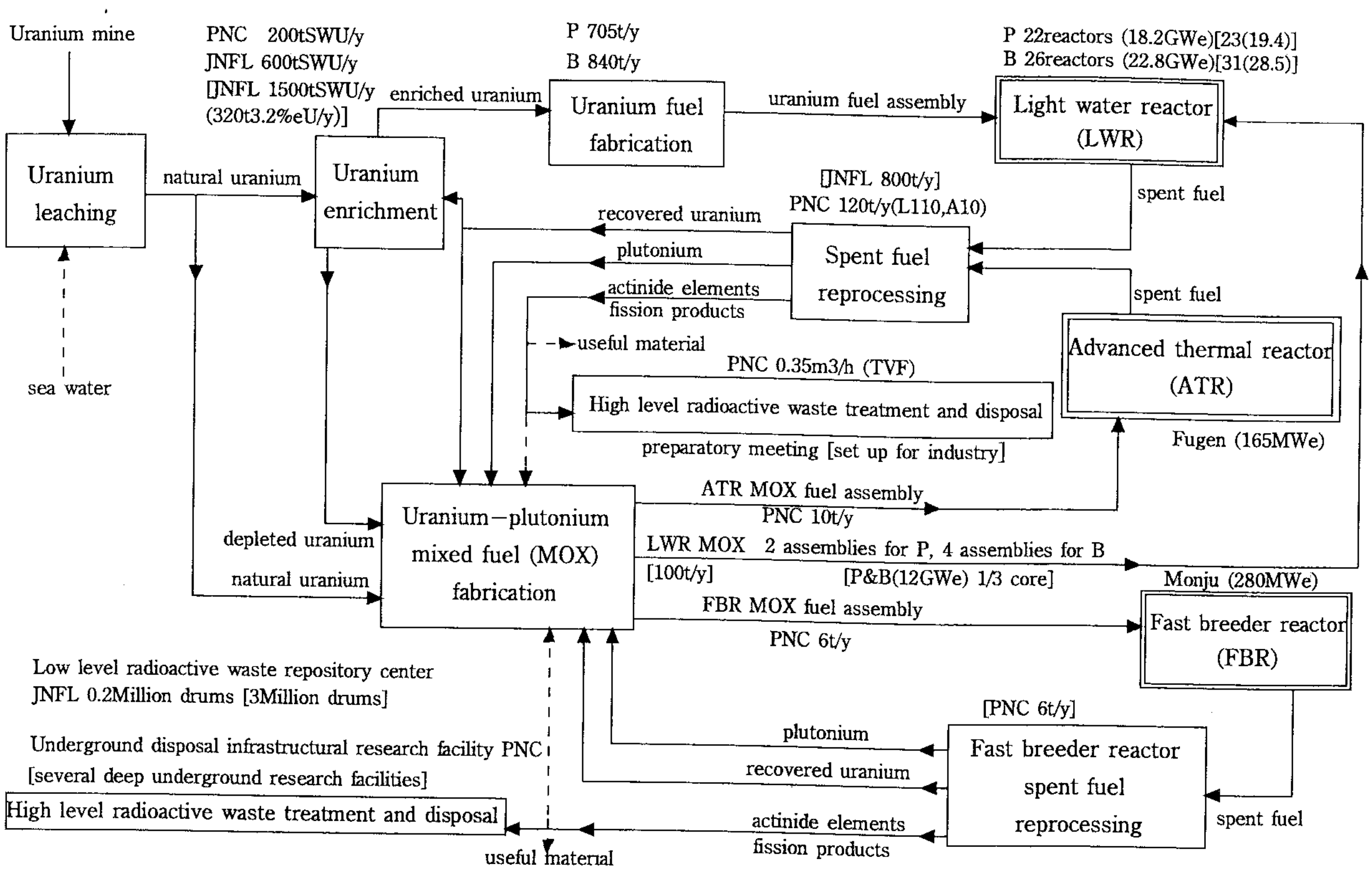

Figure 2 Nuclear Installations in Japan (1995.7)

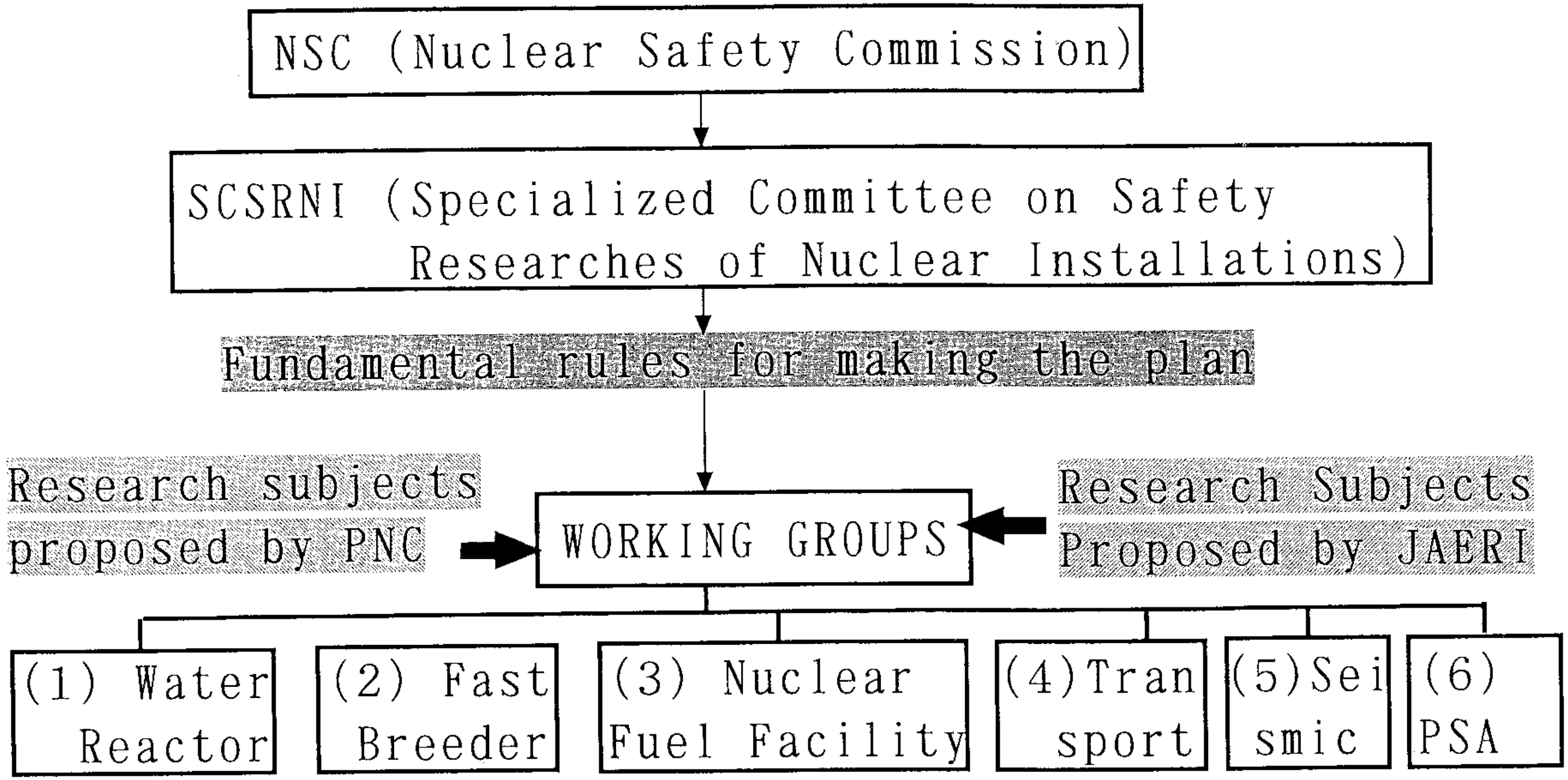

Figure 3 Set up for the 5 year annual plan

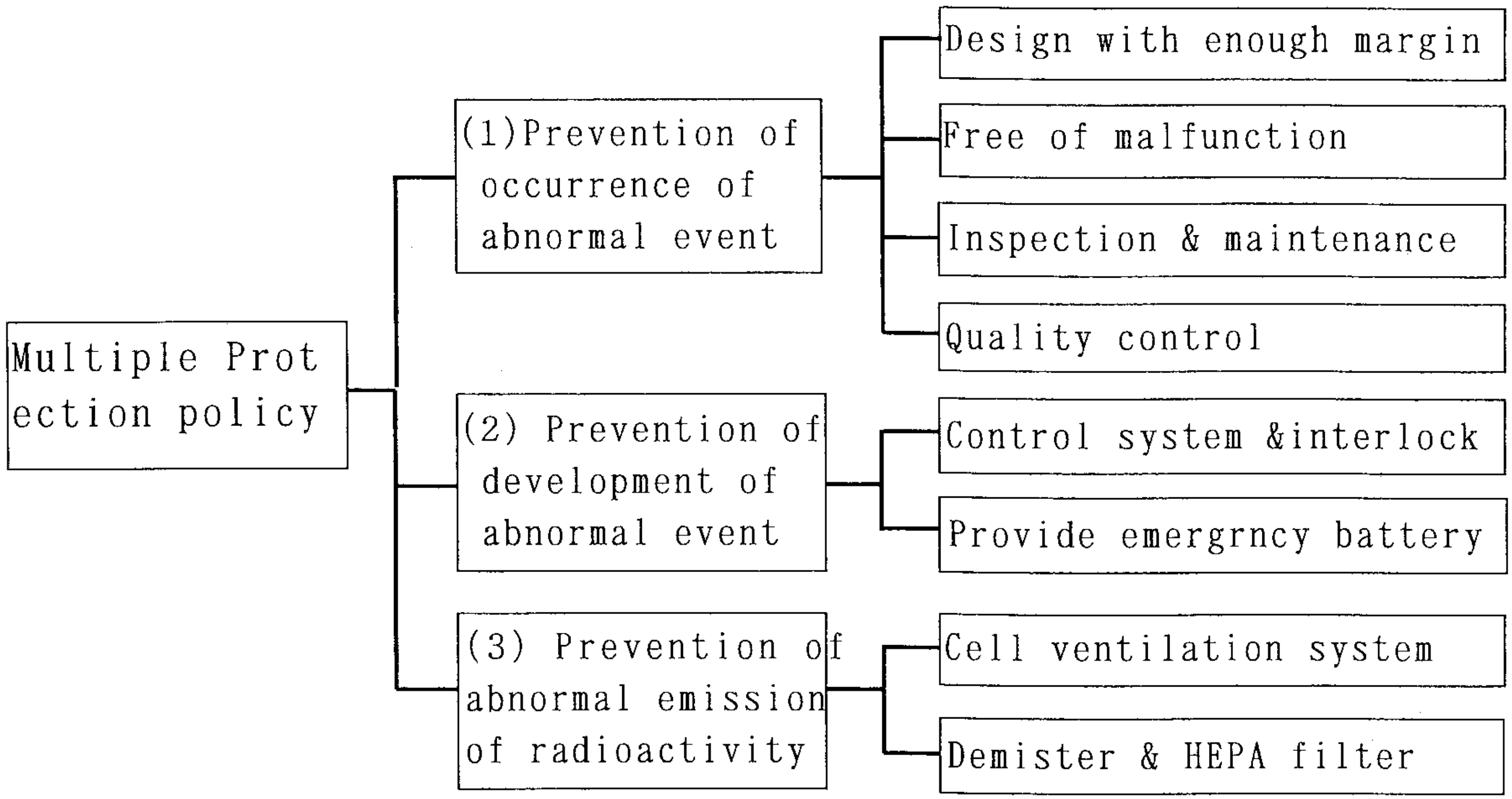

Figure 4 Defence In Depth Concept for Multiple Protection Policy

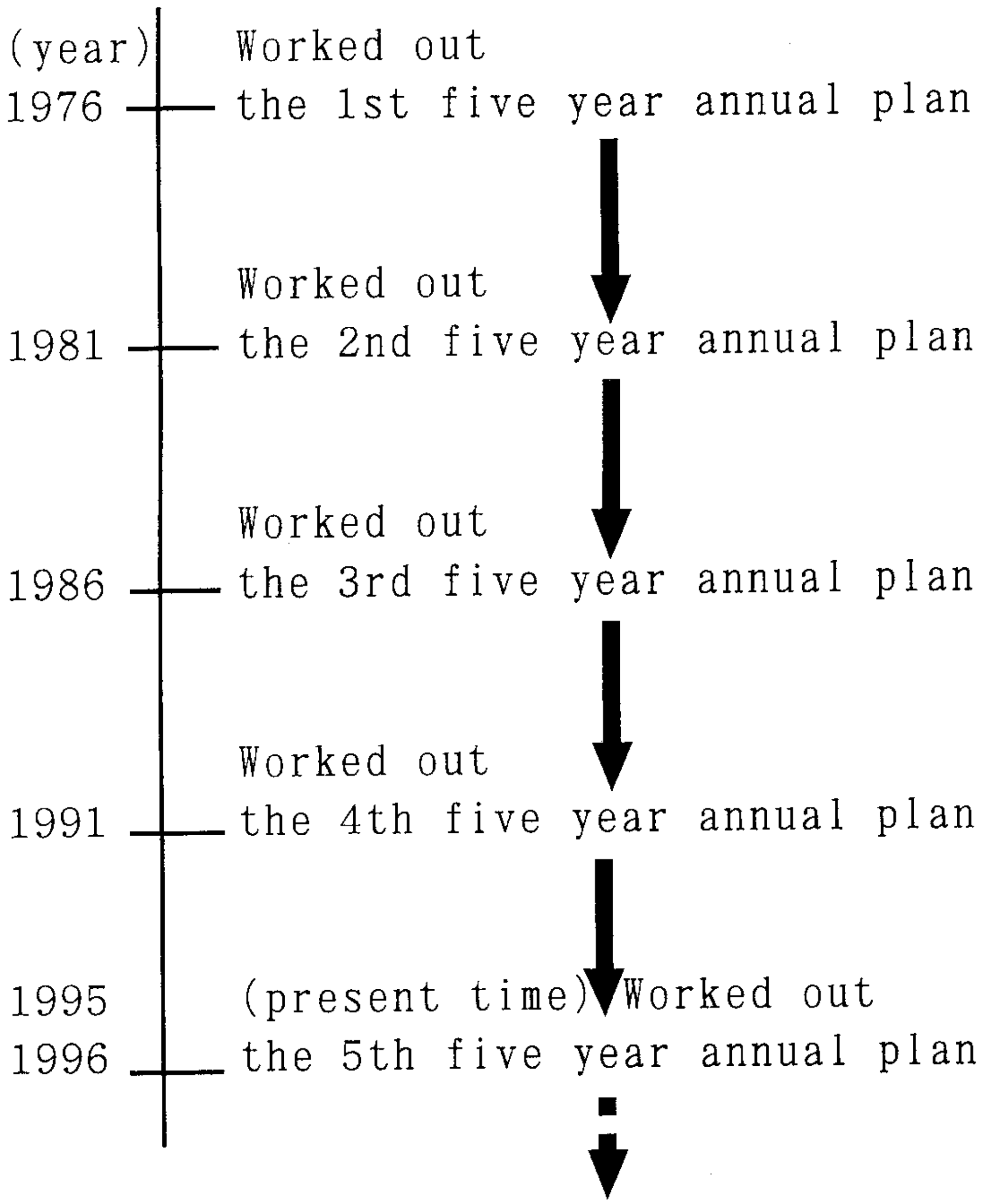

Figure 5 Chronology of the Five Year Plans for Safety Researches

Objectives of Safety Researches

1) To secure the safety of nuclear installations in preparation for the anticipated expansion of nuclear energy development and utilization.
2) To contribute to the formation of national consensus regarding the safety of nuclear installations.

Scope of Safety Researches

1) To prepare the judgement material for use in the safety criteria, guides and licensing safety reviews.
2) To improve the safety of nuclear installations.

Object Nuclear Installations

1) Water reactors
2) Fast breeder reactors
3) Nuclear fuel facilities
   (reprocessing, fuel fabrication, waste treatment)
4) Radioactive material transport casks

Figure 6 Basic Guidelines for Working Out the Annual Plan

(continued)

Requirement for the Annual PLan

1) To promote positive international cooperation.
2) To perform joint researches with universities concerning the fundamental researches
3) To make an effective use of the safety demonstration tests using real scales

Research Areas to be Focused On

1) Human factors
2) Severe accidents
3) Probabilistic safety assessments
4) Reprocessing
5) Extended aging effects
6) Reliability of digital systems

Figure 6 (continued) Basic Guidelines for Working Out the Annual Plan

Discussions about needs for safety research obtained by general survey

↓

Hearing from the national research organizations such as JAERI and PNC on each of the proposed subjects

↓

Check and review of acceptability on joint research subjects proposed by universities in Japan

↓

Selection of research subjects to be adopted in the five year annual plan judged by ;

suitability, crucialness, urgency, demands

Figure 7 Procedures for Working Out the Annual Plan

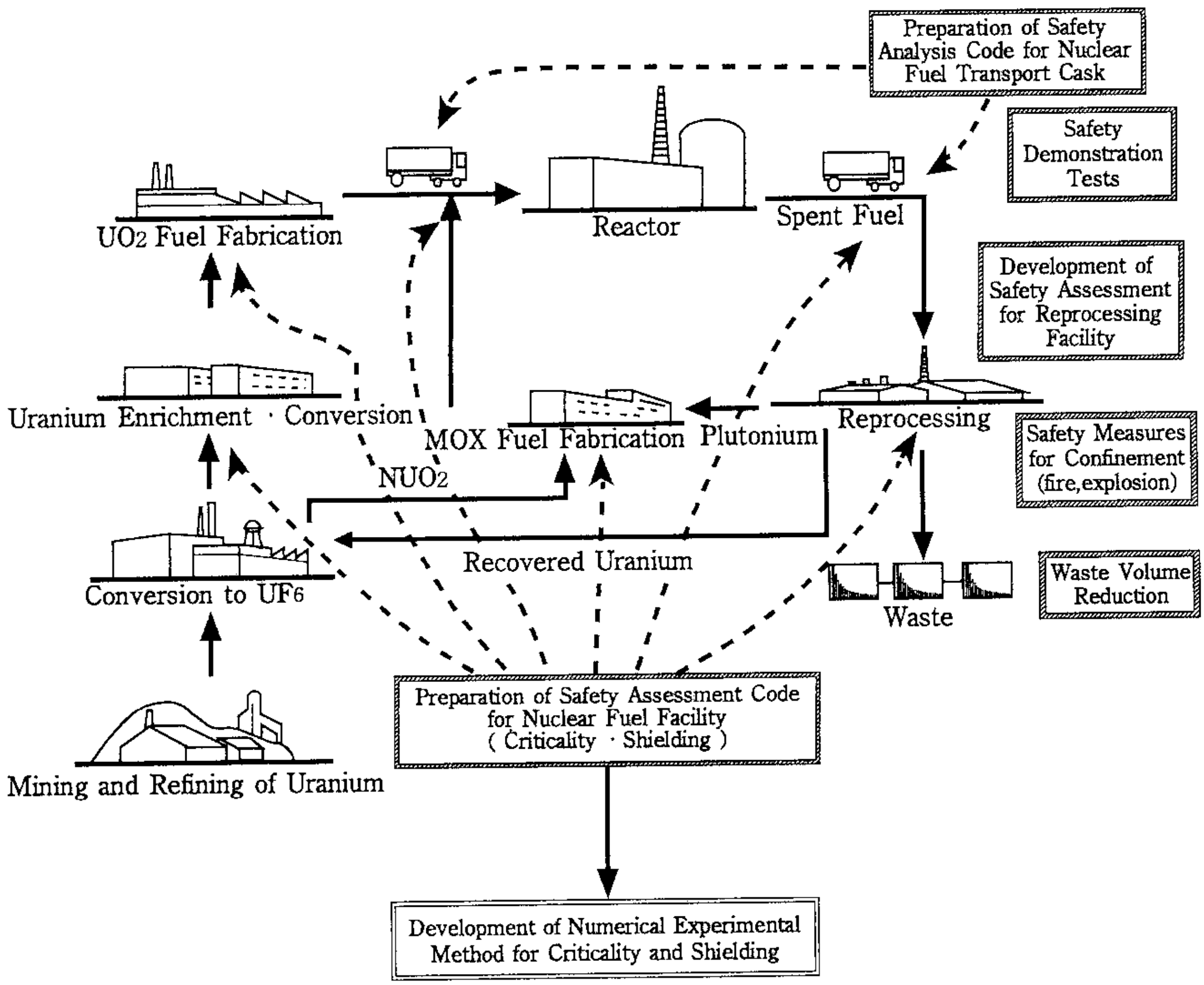

Figure 8 Examples of Relation of Each Theme of Nuclear Fuel Cycle Safety Research

STATUS AND PROSPECTS OF SAFETY RESEARCH ABOUT FUEL CYCLE FACILITIES IN FRANCE

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1. INTRODUCTION.

Although there is a good knowledge of the risks and no major accident occurred in France, as in other OECD countries, it remains useful to complete basic knowledge and to allow the quality of fuel cycle plants safety assessments to be improved further, particularly in countries equipped with a "complete" nuclear fuel cycle (France, Japan and U.K.).

The scope of the current and future IPSN (*"Institut de Protection et de Sûreté Nucléaire"*: institute for protection and nuclear safety) research deals with the whole fuel cycle. The overview presented here in NUCEF'95 symposium contains a number of specific themes, some of which have already been started. Successful conclusion of the safety researches will allow the IPSN to have a more precise understanding about specific phenomena and notably to replace "engineer judgements", though they may be based on a lot of experience and competence, by more scientifically established basic data.

The researches are done by IPSN or CEA (*"Commissariat à l'Energie Atomique"*: French Atomic Energy Commission) departments or other competent teams. The R & D uses experimental facilities operated by these bodies.

More precisely, the specific themes of research are as follows:
1) Criticality safety, which is the subject of a particular session of NUCEF'95 symposium
2) Fire hazards, which are dealt with in a particular presentation.
3) Scenarios which are particularly interesting from a safety point of view:
   - prolonged loss of cooling in concentrated fission product solution storage tanks;
   - dewatering of a spent fuel storage pond;
   - explosion risks in nuclear fuel cycle laboratories and plants;
   - dissemination of radioactive materials in case of fire in fuel manufacturing plants and in spent fuel analysis laboratories;
   - contamination transfer, including:
     - velocity limit ensuring contamination does not backscatter to openings in ventilated anti-contamination devices,
     - contamination transfer in ventilated areas,
     - re-suspension of the contamination;

- phenomenology of liquid uranium hexafluoride vaporisation into the atmosphere,
- ways and means of intervention in the event of liquid ClF_3 leakage;
- offsite explosion;

4) Seismic research.

Carring out work geared towards better understanding of basic data already resulted in improvements in safety assessments (contamination transfer mechanisms, evaluation of accident consequences).

2. RESEARCH AREAS AND PROGRAMMES.

2.1. PROLONGED LOSS OF COOLING IN CONCENTRATED FISSION PRODUCT SOLUTION STORAGE TANKS

Due to its very low probability, this sort of accident has been classified as beyond-design for spent fuel reprocessing plants. Nevertheless, this accident was studied in the frame of the determination of technical bases for the PPI ("*Plan Particulier d'Intervention*" : offsite emergency plan of public authorities) of the La Hague plant.

When such a loss of cooling happens, self-heating results in the solutions boiling until total evaporation occurs. The research involved trials on real solutions in the CEA - DPR ("*Département des Procédés de Retraitement*": process department) laboratories.

This research is now completed. The results obtained were presented in document form at international congresses on fuel cycle safety (NRC, San Diego 1990, and OECD/NEA, Tokyo 1991). They confirmed the very pessimistic character of the first global estimates.

2.2. DEWATERING OF A SPENT FUEL STORAGE POND.

This type of accident is also considered highly improbable and is classified as beyond-design. Nevertheless, the study of the consequences of dewatering in a spent fuel storage pond began in 1990. Given the increased storage capacity for irradiated material in ponds, this matter requires particular attention.

Alongside the COGEMA studies concerning the cooling of the fuel assemblies, the IPSN carries out, with CEA - DMT ("*Département de Mécanique et de Technologie*" : Mechanics and Technology Department) specific works aiming at:
- studying the storage pond dewatering phase,
- estimating the consequences of a total emptying of the pond (possible deterioration of the fuel cladding).

Generally speaking, the thermal behaviour of assemblies in the event of a total or partial dewatering involves phenomena hard to model by calculation alone.

The five-year programme, which will be completed for the end of 1995, includes:

- using existing codes to get an idea of the phenomena's order of magnitude, kinetics and time available for implementing countermeasures. This phase took place in 1991 and allowed the major parameters and corresponding R & D programme to be worked out;
- the development of codes suitable for simulating the thermal situation in the fuel assembly and the behaviour of all the assemblies in the storage pond required for a more detailed study of the transients;
- the definition of a programme of mock-up trials at scale 1 of a fuel assembly (called "PEGASE") allowing improvements to the codes to be tested and validated by comparison of the trial results with those of the calculations;
- calculations for real cases.

For 1996, a complementary parametric study program of the dependence on residual power is envisaged, to extend the validation range of the heat transfer computation module for natural convection in a square fuel rod lattice (17 x 17 type).

2.3. EXPLOSION RISKS IN NUCLEAR FUEL CYCLE LABORATORIES AND PLANTS.

a) The first subject is devoted to studying the consequences of a hydrogen explosion in the "free volume" of a fission products storage tank (sub-contracted to CEA - DMT).

The following two situations are examined:

- tank filled to its maximum (worst possible situation in terms of potential radiological effects in the case of containment burst),
- tank filled to a lower level (worst possible situation in terms of volume of exploding gas).

Calculations are in progress to evaluate the structural consequences of the explosion.

b) Recently, a programme of documentary research was started in order:

- to collate the largest amount of information available on the following:
  - inventory of the risks of explosion in "laboratories and plants", specifying the type of equipment involved and resulting damage scenario;
  - experience feedback on explosions in "laboratories and plants";
- to identify studies and research useful to assess the possibilities and consequences of such accidents.

2.4. DISSEMINATION OF RADIOACTIVE MATERIALS IN CASE OF FIRE IN FUEL MANUFACTURING PLANTS AND SPENT FUEL EXAMINATION LABORATORIES.

The following actions have been undertaken:

- documentary research on the fire behaviour of sintered or non-sintered UO_2 type material to determine useful studies and researches;
- feasibility study for fire resistance trials on spent fuel rods open at both ends in

order to study their fire behaviour and the output kinetics of fission products:

selection of representative situations,

experiment proposals: test objectives,
measured quantities,
use of results.

This would allow for a better assessment of fire risks in the following nuclear fuel cycle plants:

- fuel manufacturing plants,
- spent fuel examination laboratories.

The corresponding studies are today based on assumptions it would be advisable to check against scientifically established basic data:

- what is the fire behaviour of new, sintered UO_2 pellets stored on shelves with no specific containment before being put into the fuel rods? At what temperature do the pellets start losing their cohesion and how do the radioactive materials disperse? (First action);
- in the event of spent fuel in a fire, what degradation would it undergo? What fraction of the fuel would become pulverulent? Out of this fraction, what proportion would return to suspension in the air? What radioactivities would be discharged? (Second action).

2.5. CONTAMINATION TRANSFERS.

2.5.1 Dynamic containment at work stations.

The aims of this program are to test and specify dynamic containment techniques at laboratory and nuclear fuel reprocessing plant work stations and to evaluate the validity of current containment criteria; it will comprise two phases:

- carrying out of experimental studies, mainly in-laboratory and if possible on-site, to correlate the efficiency of dynamic containment with various operating parameters (mainly air velocity at openings, opening geometry and edge profiles, barriers, respective positions of openings and contamination sources). Digital simulations using the TRIO-EF (2D) code will be made in conjunction with these experiments;
- testing new dynamic containment techniques.

A report on literature studies (French and foreign publications) about the velocity limit guaranteeing the absence of any contamination backscattering at dynamic containment openings has been completed. From this report, it appears that there is no unequivocal evidence for the validity of the generally accepted $0.5\ m.s^{-1}$ limit.

Actually, in-situ studies in fuel fabrication plants (in particular, the FBFC unit at Romans-sur-Isère) showed the advantages of characterising dynamic containment efficiency by criteria other than the velocity at the openings of a containment cell. Local backscattering regions were found depending on the operating configurations although the velocity was $0.5\ m.s^{-1}$.

A parametric study using the TRIO-EF code for a specific configuration confirmed these results and showed, in particular, the effects of ambient ventilation conditions and of the opening dimensions on the containment backscattering coefficient.

Nevertheless, the results show that, for openings with diameters below 30 cm, the 0.5 $m.s^{-1}$ criterion appears to be justified.

To obtain a better understanding of the involved phenomena, the IPSN makes use of the following facilities:

- a test bench consisting of a containment cell with adjustable opening geometry and air inlet velocity, and instrumentation for determining gas and particle flow profiles. In particular, with this facility the air supply position with respect to the hood opening can be varied (in the lower part of the room where the hood is located, in the upper part, at the same height as the opening, scattered supply from behind a barrier), opening size, flow velocity (flow rate/surface area ratio);
- a test bench using Doppler laser anemometry for observing the velocity profile at an opening.

2.5.2 Contamination resuspension.

The factors leading to resuspension of solid or liquid contamination are, in general, poorly known. The R&D work comprises two actions:

- to gather and analyse the main data now available on this subject to provide orders of magnitude for the resuspension factors;
- to carry out experimental studies for simulating the main resuspension mechanisms liable to be encountered in the nuclear fuel cycle installations.

In this field, the significant IPSN programs are:

a) an exhaustive literature study of the resuspension and transfer factors in nuclear fuel cycle laboratories and plants.

One of the extensions envisaged for this program would be to prepare a complete state-of-the-art report for publication as a handbook.

This report would show the gaps in our knowledge and lead to establishing a future R&D program;

b) expert assessments and experimental studies of resuspension factors during the dismantling of nuclear installations. Such studies are made in order to better evaluate the potential hazards during dismantling and cleaning of radioactive cells.

2.5.3. Development of the ventilation code SIMEVENT.

The interest in using the SIMEVENT code for the safety assessment of ventilation networks was presented in document form at international congresses on fuel cycle safety (NRC, San Diego 1990, and OECD/NEA, Tokyo 1991).

The code is based on the division of the ventilation system into nodes linked by branches. An algorithm solves the mass and temperature balance equations in each node. All the components of the ventilation network are modellized, including passive (filters, ducts,...) and active (fans, ejectors,...) components, pressure and flow controllers, taking into account air leakages with other rooms or with the atmosphere outside of the buildings.

Using a given set of operating conditions as reference, SIMEVENT can be used to calculate the new operating state of the ventilation network following a simulated disturbance: mechanical (fan shutdown, hatch opening,...) or thermal (temperature increase in a room,...).

Since the first version of SIMEVENT, a steady state one, developments were validated; they concern:
- natural ventilation (in case of a total loss of electrical power),
- modelling of transfer (gases or aerosols) and filter clogging,

The validation of a version of SIMEVENT for transient states is in progress.

2.6. HAZARDS ASSOCIATED WITH LEAKAGES FROM UF_6 CANISTERS AND WITH ClF_3 FIRES.

The study entitled "phenomenology of liquid UF_6 vaporisation into the atmosphere: the case of leakage from the burst 48 Y canister valve" should first of all allow to obtain a picture of the present state of knowledge, then guide and prepare future research on the behaviour of UF_6 emitted into the environment as a liquid (particularly the fraction evaporated and the evaporation kinetics). The study will include three phases:
- documentary research on work performed in the area (theoretical, experimental and accident analyses);
- identification of the main phenomena involved in the process, phenomena modelling trials, highlighting the gaps in the knowledge;
- proposal of objectives and plans for carrying out an experimental programme allowing the gaps to be filled.

The above work is to be conducted alongside the research on hazards related to ClF_3 fires.

This latter study should allow to obtain a picture of the present state of knowledge, then guide and prepare future research on the conditions of intervention in the event of liquid ClF_3 leakage in order to limit its spread and reactions with other materials it encounters. It will include two phases:
- documentary research and synthesis of knowledge on the matter;
- identification of the phenomena involved in the reaction process between liquid ClF_6 and any material it may encounter, allowing for research proposals on intervention products.

2.7. OFFSITE EXPLOSIONS.

The objective of this program, made by IPSN in collaboration with the CEA - DMT, is to evaluate the consequences of an offsite explosion with other means than those recommended by the Basic Safety Rule ("Règle Fondamentale de Sûreté") N°. 1.2.d, by improving the accuracy of the calculations for an explosion outside in the atmosphere.

The studies started in 1993. They consisted in applying the PLEXUS code to a situation already studied by IPSN with the method suggested by the Basic Safety Rule: explosion of a cloud from the evaporation of a layer of liquid hydrogen in air under detonation or deflagration conditions (evaluation of the overpressure produced as function of source distance, in the atmosphere without barriers).

These studies will be continued in the following directions:
- effects of cloud shape,
- effects of barriers,
- experimental study of mechanical effects of shock waves on barriers,
- studies of other gases.

The results will be applied to a real case on site, involving the presence of an obstacle.

2.8. SEISMIC RESEARCH.

For nuclear power plants, IPSN developed a method for determining the earthquake intensity to be taken into account for the design of the plants; this method was also applied to the nuclear fuel cycle plants. It consists of historical and geological studies to assess the intensity of what is termed the "maximum historically probable earthquake", the earthquake taken into consideration in the design basis then being the "maximum safety earthquake" defined as being one degree higher than the maximum historically probable earthquake in the MSK scale. Response spectra with a number of damping values are then established by using a collection of data on recent earthquakes from different regions of the world. A permanent updating of the data is ensured.

Complementary studies were started in the area of plant earthquake resistance. Spanning six years, the programme's objectives are to better assess the existing margins concerning the structures' earthquake resistance in the event of an earthquake. The studies started with an initial phase of static trials on solid walls and then of dynamic trials on walls either solid or with centred or non-centred openings. The next phase of trials looked at the effect of the number of openings in a wall and on the effect of transverse walls on overall resistance. Generally speaking, the trials did not demonstrate any clear-cut resistance of the buildings but they did allow calculation methods to be developed and applied to the earthquake resistance of real structures. These studies are sub-contracted to CEA - DMT.

3. CONCLUSION.

The safety of a nuclear installation is primarily the responsability of its operator. As an independant expert, the French IPSN performs research works to better assess the operators' proposals.

The foregoing shows that IPSN is carrying out works in all the main fields concerning the safety of plants and laboratories of the nuclear fuel cycle.

THE REGULATORY PROCESS, NUCLEAR SAFETY RESEARCH AND THE FUEL CYCLE IN THE UNITED KINGDOM

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CONTENTS

INTRODUCTION

1. The main legislation governing the safety of nuclear installations in the United Kingdom is the Health and Safety at Work Act 1974 (HSWA) and the associated relevant statutory provisions of the Nuclear Installations Act 1965 (as amended). The HSWA sought to simplify and unify all industrial safety legislation and set in place the Health and Safety Commission (HSC) and its executive arm, the Health and Safety Executive (HSE). The Health and Safety Executive's Nuclear Safety Division (NSD) regulates the nuclear activities on such sites through HM Nuclear Installations Inspectorate (NII).

2. Under the Nuclear Installations Act (NIA) no corporate body may use any site for the purpose of installing or operating any reactor, other than such a reactor comprised in a means of transport, or other prescribed installation unless the operator has been granted a nuclear site licence by the Health and Safety Executive. Nuclear fuel cycle facilities are examples of such prescribed installations.

3. Each nuclear site licensee is responsible for all aspects of safety on the site. To aid in the regulation of safety on nuclear licensed sites, conditions are attached to each licence. One of the conditions requires the preparation of a safety case to justify the safety of activities on the site. Nuclear safety research is seen as a key element in maintaining up-to-date knowledge supporting the safety case. Such research may be site specific or generic. This paper addresses how the NSD ensures that, through the HSC's Co-ordinated Research Programme, adequate consideration is given to generic nuclear safety research. Site specific research is a matter for the licensee as part of the normal regulatory process.

4. The development of NSD's Nuclear Research Index (NRI) will be described and how the Index has been applied to reactor safety issues within HSC's Co-ordinated Research Programme.

5. Research in the nuclear fuel cycle sector is excluded from HSC's Co-ordinated Research Programme. However NSD recognises that the fuel cycle industry has been carrying out detailed nuclear safety research over many years in addition to research in support of commercial opportunities. A Nuclear Chemical Plant Research Index (NCPRI) has been developed by NSD similar in structure to the NRI . The NCPRI is intended to provide a focus for discussions with fuel cycle licensees on nuclear safety research matters.

6. As a regulator, the NSD must be satisfied that an adequate level of safety research is maintained to justify the continued safe operation of nuclear plant. Such activities as the development of the NRI and NCPRI help to achieve that aim.

THE STRUCTURE AND ROLE OF NSD

7. HSE's mission is to ensure that risks to people's health and safety from work activities are properly controlled. HSE consists of 11 divisions (Figure 1) of which the Nuclear Safety Division is one. NII is the operational arm of NSD and the divisional structure is shown in Figure 2.

8. The Aims of NSD are:

a. To secure the maintenance and improvement of standards of safety at civil nuclear installations

and

b. To secure the protection of workers and members of the public from ionising radiations.

9. The duty of NSD is to see that the appropriate standards are developed, achieved and maintained by the licensee, to ensure that any necessary safety precautions are taken and to monitor the safety of plant by means of its powers under the licence and relevant regulations. This duty is carried out by establishing safety requirements for the protection of both workers and members of the public and by inspection for compliance with these requirements at all stages from design and construction to operation and eventual decommissioning

10. NSD's six Branches are organised in a system of matrix management.

11. The two Policy Branches provide help and information to government ministers and their principal advisers, advice to the Health and Safety Commission, nuclear licensing policy, co-ordination of NSD's international work and formulation of radiation protection policy for HSE. These Branches also provide management functions for NSD's external research support, training and other management services including secretariat to a number of advisory committees. The research policy function links to the HSC's Co-ordinated Research Programme and the industry to ensure suitable management arrangements are in place.

12. The two Inspection Branches administer the site licenses for nuclear power stations and other nuclear sites. Important features of their work include checking for compliance with site licence conditions and other regulations dealing with radiation protection matters. Site visits and associated activities form a major part of the work.

13. The two Assessment Branches support the inspection and policy branches providing scientific and technical advice and assessments, usually of licensees' safety cases. The areas covered are shown in Figure 2. Broadly, Branch B is assigned scientific topics and Branch C engineering topics, although that is an over simplification. Representatives from these Branches provide the Division's technical input to nuclear safety research.

14. NSD, as part of HSE, is well positioned to meet the demands of a changing industry and a changing environment. There is in place a well established and recognised goal-setting regulatory regime which can deal with new technologies and new industrial structures. The integrated organisation of NSD allows an effective response to new issues which may have repercussions for research. For example, the UK government announced in May of this year its intention to restructure the nuclear power industry and privatise the more modern power stations. To ensure that the high safety standards of the industry are maintained all the affected nuclear power station sites are being relicensed. Part of the new licence applicants' submissions have to address safety related research. NSD is assessing such submissions.

15 Notwithstanding NSD's regulatory role, duties and functions, the licensee is solely responsible for nuclear safety on each licensed site.

NUCLEAR SITE LICENCE

16. The heart of the regulatory control system on nuclear licensed sites is the licence (and its attached conditions) which can only be granted to a body corporate to use a site for carrying out specified activities. Licensing applies through the lifetime of an installation including decommissioning. The overall form and structure of the licence is the same for all nuclear installations. The licence defines the body corporate who is licensed and in schedules attached to that licence are definitions of the processes permitted, the site location and a series of licence conditions (Appendix I).

17. Guidance on the legal requirements and the duties and responsibilities of licensees is available in the form of the recently published HSE document "Nuclear Site Licences: Notes for Applicants".

18. Every licence is unique in detail to its site and may be varied to exclude any part of the site the licensee no longer requires for licensable activities. Before making such a variation the NSD, as the administrator of the licensing function, must be satisfied that there is no danger from ionising radiations from anything on that part of the site to be excluded.

19. A licence may be revoked by the NSD or surrendered by the licensee. In either event, the corporate body which was granted the licence will still remain responsible for the safety of activities on that site. This period of responsibility ends when a new licence has been granted, either to the same licensee or other body, or NSD gives written notice that, in its opinion, there has ceased to be any danger from ionising radiations from anything on that site.

20. The NSD can, at any time, attach to a licence conditions which appear necessary or desirable in the interest of safety. There is also the power to revoke conditions attached to a licence allowing the licence to be tailored to specific circumstances and the phase in an installation's life. The present thirty five licence conditions have evolved with the aim of producing consistent safety requirements which are goal-setting and flexible. These are listed in Annex 1. They deal with such matters as:

a. marking the site boundary

b. appointment of "Duly Authorised Persons"

c. the production of safety cases

d. incident reporting

e. operating rules, limits and conditions

f. design, modification, operation, decommissioning and maintenance

g. control, supervision and training for staff

h. control of waste (without prejudice to the legislation administered by the Department of the Environment relating to waste disposal)

i. emergency arrangements.

21. In the main, these licence conditions require the licensee to make and implement adequate arrangements appropriate to that condition. Therefore, each licensee can develop arrangements that best suit its business whilst demonstrating that safety is being adequately managed. Although the conditions are the same for most licensees the arrangements to comply may change as the installation changes from one phase to the next and, indeed, change from licensee to licensee. By this process the licensee is responsible for the application of detailed safety standards and safe procedures for the installation.

22. The licensee's arrangements are elements of an overall safety management system. NSD reviews these arrangements to ensure they are clear, unambiguous and adequately address the main safety matters. In particular, NSD looks for consistency between the assumptions and the commitments made in the safety case and the safety management system. The licensee will need a level of resource commensurate with the risk. A management system should demonstrate a commitment to health and safety including a definition of:

a. lines of authority to adequately control the licensee's own personnel and contractors

b. adequate staff

c. precise definition and documentation of duties

d. integration of Health and Safety into jobs

e. suitably qualified and experienced persons ensuring adequate in-house expertise

f. provision of or access to, high levels of safety expertise used actively in peer review and safety audit

23. There is no specific reference to research in the licence conditions but the need to prepare a safety case (Annex 2 -Licence Condition 14 - Safety Documentation) implies an understanding of the technology involved. This requirement for a safety case necessitates the licensee understanding the basic science of the processes involved as far as it affects the safety of operations. The licensee may acquire this understanding in a variety of ways but generally this is by carrying out, commissioning an outside agency to carry out or gaining access to the necessary fundamental research.

RESEARCH

24. There are many factors to be considered in the field of nuclear safety research including the interfaces between regulators, operator/licensees and contractors. This paper considers the role of NSD as the regulatory body for nuclear safety and the interfaces established to ensure the adequacy of nuclear safety research in the United Kingdom. Of particular interest is the relation with the industry and the procedures that are developing to ensure an appropriate level of nuclear safety research is maintained. Factors which will be considered include the identification of research needs, the procurement of such research and the maintenance of research capability.

25. In relation to research activities the" nuclear industry " in the United Kingdom is best considered as two separate sectors . These are the civil thermal reactor sector and the fuel cycle or chemical plants sector. This split is important as, historically, separate approaches to the control of research were adopted in the two sectors. There has been significant central government involvement in reactor research matters because of the general policy in the past to develop nuclear electric generating capacity. The research needs for nuclear chemical plants were not perceived as so much in need of central control and the licensees had more freedom to establish their own research agenda.

26. As identified above there is a licence condition dealing with the need to have in place arrangements for the preparation of a safety case to demonstrate the safety of activities carried out on a nuclear licensed site. The preparation of the safety case requires the licensee to have a comprehensive understanding of the hazards and risks associated with the operations on the site so they can continue to be carried out safely. NSD considers research as a key element in the understanding of the hazards and risks.

27. Such research could address any features that may have an impact on operations including process, facility and equipment, ageing, process materials deterioration , operator interfaces, instrumentation up-grades and waste and decommissioning.

28. The responsibility for generic nuclear reactor safety research was transferred from the Department of Trade and Industry to HSC/HSE in 1990 under arrangements agreed with the President of the Board of Trade , the government minister with overview responsibilities for the nuclear industry. This resulted in the development of the HSC Co-ordinated Research Programme based on agreed Guidelines (Annex 3) which identify primary objectives to be met and other factors to aid in the control of research. This transfer of responsibility allows nuclear safety research to be monitored transparently by the regulatory body and removes any potential conflict of interests as a result of industry privatisation initiatives. The HSC Co-ordinated Programme specifically excludes fuel cycle research and the Guidelines are specific to reactor systems but experience gained in this area is valuable in developing arrangements with fuel cycle licensees.

29. As revised systems have been developed in the reactor sector NSD was conscious of the need to improve interfaces with the fuel cycle licensees on generic research matters. These interfaces were not as well defined as for the reactor sector but with these new initiatives some of the uncertainties will be removed. The structure of the fuel cycle sector will result in the detail of the arrangements being different to those for the reactor sector but with the aim of achieving a greater degree of transparency for relevant research areas.

HSC CO-ORDINATED RESEARCH PROGRAMME

30. To ensure the UK government has a view on the requirement for applied nuclear safety research the HSC Co-ordinated Research Programme has been established to influence the level and topics of such research. With the further development of the reactor sector into a commercial environment this independent view on nuclear safety matters is seen as essential to ensure a balanced programme of nuclear safety research continues to be carried out . This responsibility is exercised through the HSE, who has delegated day to day control to its Nuclear Safety Division.

31. The HSC Programme comprises three major elements:-

i) The Industry Managed Collaborative Programme.

ii) The Industry Direct Programme

ii) The HSE Levy Programme;

but excludes a number of activities :-

i) nuclear safety research related to the fuel cycle;

ii) individual or collaborative research undertaken by licensees for purposes other than safety, or research specific to a site;

iii) nuclear safety research commissioned by NSD for specific licensing decisions which may apply today or in the future.

32. A well developed strategy has been prepared for handling nuclear reactor safety research involving the industry and the NSD and this will be described in some detail as arrangements for nuclear chemical plants have to be finalised. The main objectives of the strategy are:-

- to demonstrate how an adequate and balanced programme of applied research will continue to contribute to improved standards of safety and address NSD's needs;

- to identify areas where capability is retained or access to expertise needs to be secured over a defined period;

- to identify and account for changes in the environment in which the Programme operates:

- to foresee changes in these priorities over the next 5 years;

- to satisfy internal management and planning Objectives.

33. Key functional components of the system are the Nuclear Research Index (NRI), the Industry Management Committee (IMC) and associated Technical Working Groups (TWGs). A Steering Group comprising senior members of the HSE management, major reactor sector licensees and external representatives from other government departments and consultative bodies has been formed. This Group is required to have an overview of the Programme ensuring that it is balanced and agreeing its forward strategy.

35. The funding of the HSC Co-ordinated Programme is by a levy on those parts of the industry who are likely to benefit from the research or are seen as promulgating the need for such research. This funding is a part of the costs of regulating the nuclear industry which are chargeable to the industry under the present legislation. The overall current budget for this Programme is £23.5M

NUCLEAR RESEARCH INDEX

36. The NRI has been developed as a means by which NSD identifies those nuclear safety issues which need to be addressed in a generic research programme. These issues are generated as a result of NSD's inspection and assessment regulatory activities. The aim of each issue is to make a clear statement on the particular concern so that the issue can be cleared by a specific research task. This may not always be possible and a number of parallel or consecutive research contracts may have to be placed to clear specific issues.

37. The NRI is divided into a number of discrete Sections (Annex 4). Each issue is allocated to the relevant section and eventually considered by the relevant Technical Working Group. Site specific concerns are excluded from the NRI as these are the responsibility of the respective licensee. This recognises that the licensee has the responsibility for the safety associated with an operational licensed site and is expected to commission adequate research to clear such matters. The generic issues are those which are not specific to one site but may have an impact across a number of sites or licensees or in some circumstances across different reactor types. The current NRI contains of the order of 290 issues some of which are being addressed in the 1995/96 Programme and others which may form research contracts in future years.

38. The NRI is an NSD document and the responsibility for reviewing and up-dating rests with NSD and is carried out each year. This allows new issues to be added , progress on clearing existing issues to be identified and a record to be maintained of those issues which have been completed. The document gives a structure to the workings of the various Technical Working Groups and the Industry Management Committee and provides a focus for the various technical areas in the reactor sector.

TECHNICAL WORKING GROUPS

39. Each section of the NRI has a specific Technical Working Group (TWG) which has members representing those parts of the industry contributing to that technical area and representatives from NSD. Specific aims of these groups include considering the issues identified in the relevant section of the NRI, developing research proposals to address each issue and monitoring the contracts placed. In addition the TWGs have members who are experts in the field and are able to advise on the further development of issues which may be of interest to the NSD. TWG members are also expected to consult with relevant outside individuals, bodies and agencies who have knowledge of the topics involved or may be suitable contractors. Generally the TWGs are administered by industry representatives who take the lead in preparing contract specifications, tender documents and placing the final contracts. This is seen as achieving a stated aim in that the nuclear industry is seen to be procuring research that satisfies NSD's views on its needs for safety research.

INDUSTRY MANAGEMENT COMMITTEE

40. The Industry Management Committee (IMC) was formed to allow the industry to manage its part in taking responsibility for nuclear safety research. A key function of this committee is to co-ordinate the separate elements of the Technical Working Groups into a cohesive research programme, the Industry Managed Collaborative Programme, which is

open to review and monitoring by the HSC. When HSC has ratified the proposed programme the IMC has the further responsibility of ensuring the efficient procurement of the necessary research. A consequence of the arrangements is that the intellectual property rights for research conducted under this programme belong to the members of the IMC. This allows the industry to make effective use of the results of such research and to exploit the benefits of the research in interactions with other reactor operators. The IMC is comprised of representatives of those licensees who contribute to civil power reactor research activities.

41. Fuel cycle licensees are not members of the IMC as the present system was developed from central government arrangements for reactor systems and fuel cycle research was specifically excluded from the HSC Programme.

42. In addition to the Industry Managed Collaborative Programme, which has a current budget of £8.7M, the licensees have their own programmes of research. These Industry Direct Programmes form part of the HSC Co-ordinated Programme and the current budget is £12.7M. Agreements have been reached which allow HSE access to these programmes so that they can be taken into account when the HSC Co-ordinated Programme is being defined.

THE HSE LEVY PROGRAMME

43. This Programme has three main elements:-

i) nuclear safety research issues within the scope of the HSC Co-ordinated Programme but not adequately addressed in the IMC Programme;

ii) research commissioned to maintain sources of independent advice to HSE;

iii) international research collaboration which requires the participation of the UK Government.

44. This part of the HSC Co-ordinated Programme is not managed by the nuclear industry but by HSE directly. The financing of such research is obtained by levying a charge on the industry to cover the component parts of the Programme. Over the years this Levy Programme has been reducing as a main thrust of the HSC Co-ordinated Programme is to make the industry take responsibility for the provision of nuclear safety research. A large Levy Programme is not consistent with that aim. The current budget for the Levy Programme is £2.1M.

NUCLEAR CHEMICAL PLANTS RESEARCH

45. For historical reasons research carried out in the fuel cycle industry has not been subject to the same level of scrutiny by HSE as is now in place for power reactors. NSD has monitored fuel cycle research as it has been associated with a particular projects but there was no overall monitoring of each licensee's programme. As a development of NSD's greater involvement with research matters, discussions have been progressing with the individual fuel cycle licensees to conclude a research agreement which will provide NSD with access to generic safety related research funded by the licensees.

46. Research carried out by the fuel cycle licensees is known to be extensive and includes research carried out for commercial reasons. The separation of research into categories which may be useful to NSD could be difficult and hence our interactions with the licensees will be based on an index , the Nuclear Chemical Plants Research Index (NCPRI), prepared by NSD. This Index will be the focus for information exchange. The Objective of the NCPRI is to identify issues against which the licensee can provide information. The NCPRI has been sub-divided into the same generic Sections as the NRI to allow easy cross referencing of issues and hence aid in reducing the duplication of effort.

47. The structure of the fuel cycle industry may require separate interface arrangements with the NSD to acknowledge the commercial interests of the different organisations and provide for suitable confidentiality. This protection for the licensees is a direct result of the commercial nature of the industry.

48. NSD recognises that the fuel cycle industry has been carrying out detailed safety research over many years in addition to research in support of commercial opportunities. Information exchanged using the NCPRI as a focus will make this research more transparent and allow NSD to monitor and gain confidence in the level of nuclear safety research carried out by fuel cycle licensees.

FUTURE DEVELOPMENTS

49. Research is an essential part of the nuclear industry's activities but there are a number of factors which could influence the level of research undertaken in the future. These factors include competition within the industry, maturing technology, a reducing level of government control and influence, industry re-organisation and profitability.

50. As a regulator, the NSD must be satisfied that an adequate level of safety research is maintained whatever the circumstances of the industry to justify the continued safe operation of nuclear plant. Activities such as the development of the NRI and NCPRI help to achieve that aim.

COLLABORATION

51. As mentioned above there are a number of influences which are requiring both the industry and the regulator to question established practices and ways of working. Financial pressures both on the industry and the regulators are requiring systems which are transparent and demonstrate value for money. One method of achieving this aim is to collaborate with other agencies both in the UK and overseas on the transfer of information including research data. In addition the NSD has secured, or is attempting to secure, collaborative research agreements with relevant overseas agencies, for example such an agreement was signed with USNRC in April 1995, one with France's IPSN is imminent and similar protocols are being sought with other European regulators. There are a number of generic safety areas which allows such collaboration to benefit all parties involved covering both the reactor and fuel cycle sectors. Such collaboration may take the form of bilateral agreements or be organised centrally by a suitable agency e.g. the European Union , IAEA or OECD/NEA. Such arrangements are beneficial in many ways but it is important that close control is maintained to ensure that the intended aims of such agreements continue to be met.

ACKNOWLEDGEMENT

52. This paper has drawn freely on the work of a number of colleagues. I would like to thank many colleagues in the Division for their help in preparing this paper. The views expressed are those of the author and are not necessarily the views of the NSD.

Annex 1

LIST OF STANDARD LICENCE CONDITIONS

No	Title
1	Interpretation
2	Marking of the Site Boundary
3	Restriction on Dealing with the Site
4	Restrictions on Nuclear Matter on the Site
5	Consignment of Nuclear Matter
6	Documents, Records, Authorities and Certificates
7	Incidents on the Site
8	Warning Notices
9	Instructions to Persons on the Site
10	Training
11	Emergency Arrangements
12	Duly Authorised and Other Suitably Qualified and Experienced Persons
13	Nuclear Safety Committee
14	Safety Documentation
15	Periodic Review
16	Site Plans, Designs and Specifications
17	Quality Assurance
18	Radiological Protection
19	Construction or Installation of New Plant
20	Modification to Design of Plant Under Construction
21	Commissioning
22	Modification or Experiment on Existing Plant
23	Operating Rules
24	Operating Instructions
25	Operational Records
26	Control and Supervision of Operations
27	Safety Mechanisms, Devices and Circuits
28	Examination, Inspection, Maintenance and Testing
29	Duty to Carry Out Tests, Inspections and Examinations
30	Periodic Shutdown
31	Shutdown of Specified Operations
32	Accumulation of Radioactive Waste
33	Disposal of Radioactive Waste
34	Leakage and Escape of Radioactive Material and Radioactive Waste
35	Decommissioning

Annex 2

LICENCE CONDITION 14. SAFETY DOCUMENTATION

(1) Without prejudice to any other requirements of the conditions attached to this licence the licensee shall make and implement adequate arrangements for the production and assessment of safety cases consisting of documentation to justify safety during the design, construction, manufacture, commissioning, operation and decommissioning phases of the installation.

(2) The licensee shall submit to the Executive for approval such part or parts of the aforesaid arrangements as the Executive may specify.

(3) The licensee shall ensure that once approved no alteration or amendment is made to the approved arrangements unless the Executive has approved such alteration or amendment.

(4) The licensee shall furnish to the Executive copies of any such documentation or any such category of documentation as the Executive may specify.

Annex 3

GUIDELINES ON NUCLEAR SAFETY RESEARCH AND RELATED WORK ISSUED BY THE PRESIDENT OF THE BOARD OF TRADE TO THE HEALTH AND SAFETY COMMISSION

These guidelines amend and update those issued with effect from 1 April 1990 regarding nuclear safety research responsibilities which are, or will be, managed on the Commission's behalf by the Health and Safety Executive (HSE). They take effect from 1 April 1994 except where otherwise indicated.

The guidelines take into account the fact that the safety of nuclear installations is the responsibility of the licensees of such installations, and that research covering their safety is largely undertaken or commissioned by commercial licensees, if necessary at the request of the HSE's Nuclear Installations Inspectorate (NII). They also take account of the public interest in maintaining the availability of the research capability needed for regulatory purposes.

1. Primary Objectives

i) To ensure that adequate and balanced programmes of nuclear safety research continue to be carried out, based on a view of the issues likely to emerge both in the short and long term.

ii) To ensure that, as far as reasonably practicable, the potential contribution which such research can make to securing higher standards of nuclear safety is maximised.

iii) To ensure that the results of any such research having implications for nuclear safety are disseminated as appropriate.

2. Supporting Objectives

i) To take account of the desirability of maintaining a sufficient range of independent capability to ensure the attainment of the primary objectives.

ii) To ensure that proper account is taken of the advantages of international collaboration in furthering the primary objectives.

3. Research and related work covered by HSC Co-ordinated Programme

These arrangements cover research and related work which has as a primary purpose the improvement of nuclear safety, offers a potential return in terms of greater safety standards at reasonable cost, and which is relevant to any activity or process associated with operation or decommissioning of nuclear power systems on a UK licensed site.

The HSC Co-ordinated Programme may also include work which:

i) would not be undertaken by commercial licensees on their own account;

ii) though of potential interest to commercial licensees is more appropriate for HSC to retain the proprietary rights;

iii) though required by licensees, may for legal or contractual purposes require HSC to retain the proprietary rights (eg. where government participation is required for collaboration with other countries).

4. Research not covered by HSC Co-ordinated Programme

i) Research which is undertaken individually or in collaboration by, or on behalf of, commercial licensees primarily for purposes other than safety; or to meet licensing conditions or their own safety design rules;

ii) Research which is commissioned by the NII to enable it to take specific licensing decisions, or by the licensees as a particular condition of their licences.

5. Determination of HSC Co-ordinated Programmes and Budgets

The HSC should determine the programmes in the light of consultations by HSE with:

i) the nuclear industry bodies concerned;

ii) the Advisory Committee on the Safety of Nuclear Installations (ACSNI);

iii) such other sources as they consider appropriate;

and upon recommendation by the HSE.

6. Basis for Cost Recovery

i) The costs of research, and its management by HSE, should be recovered from licensees of, or licence applicants for, nuclear installations to the safety of which the research appears to the HSC/HSE to relate.

ii) They should be recovered in proportions which:

- reasonably reflect the costs of research (and its management by HSE);

and which take account inter alia:

- in the case of existing licensees, of the scale on which each of them undertakes the activities to which the research relates.

- in the case of licence applicants, of the scale on which each of them plans to undertake the activities to which the research relates.

The scaling factors will be specified in a Memorandum of Arrangements between HSE and the licensees.

7. Proprietary Information

In exercising their co-ordinating role, the HSC/HSE should use their best endeavours to protect any proprietary information which comes to their attention, so far as this is consistent with the requirements of nuclear safety.

8. The Department of Trade and Industry's residual Programme of Safety Research

The Department of Trade and Industry may retain responsibility for funding certain safety research which is more relevant to its own responsibilities than to those of the HSC/HSE. The results of this work will nevertheless be made available to HSC/HSE, and their views may be sought on its content and direction.

Annex 4

SECTIONS OF THE NUCLEAR RESEARCH INDEX

Plant Life Management - Steel Components

Plant Life Management - Civil Engineering

Plant Life Management - Ageing of Mechanical and Electrical Components

Chemical Processes

Fuel and Core

Fission Products

Reactor Physics, Criticality and Shielding

Heat Transfer and Thermal Hydraulics

Severe Accidents and their Management

External Events

Control and Instrumentation

Human Factors

Probabilistic Safety Assessment

Radiological Protection

Decommissioning and Radioactive Waste

Figure 1 : Health and Safety Executive

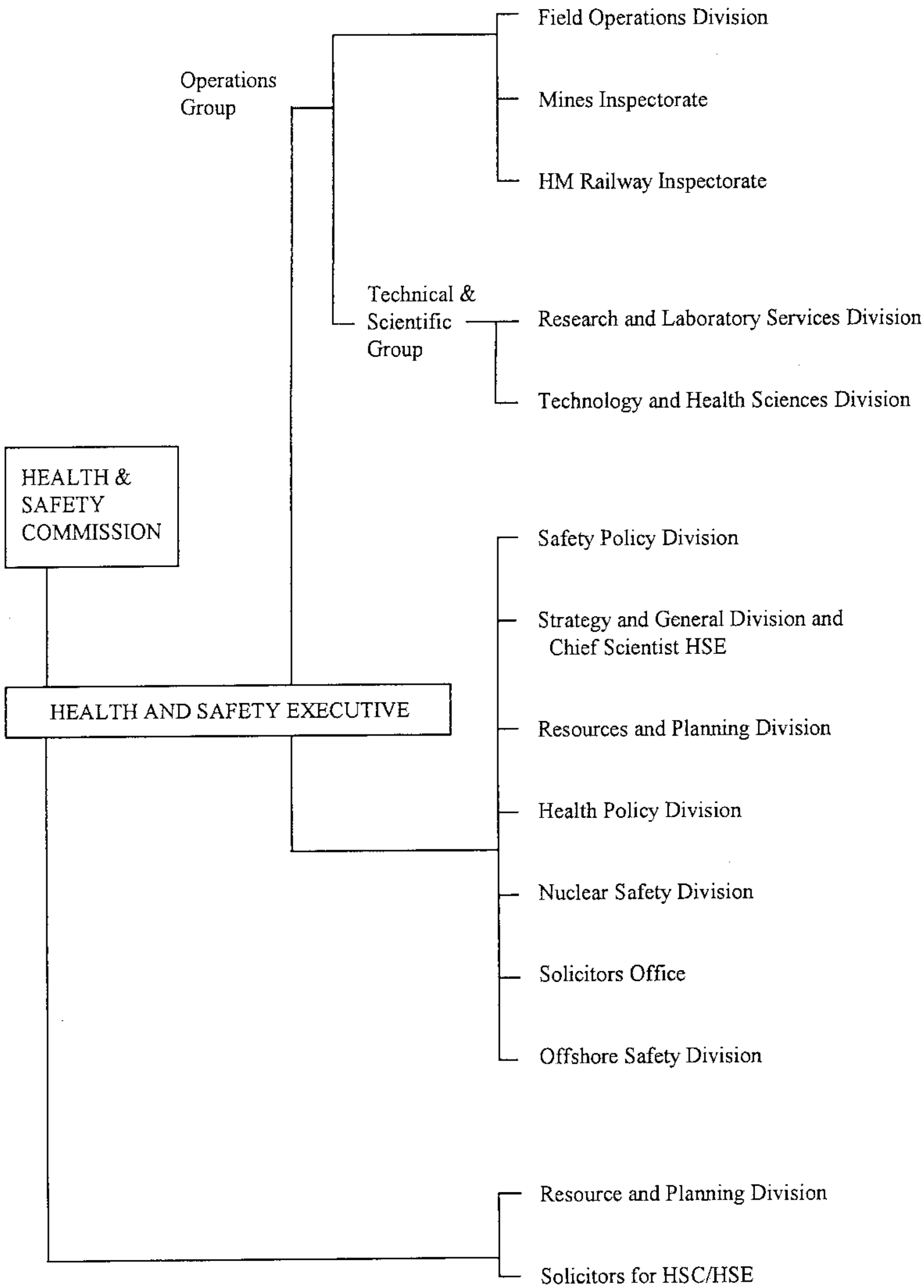

Figure 2 : Nuclear Safety Division

- CI(NI)
  - BRANCH A POLICY STRATEGY AND INTERNATIONAL
    - UNIT 1 General Policy and Planning, Information Systems
    - UNIT 2 International Policy
  - BRANCH B ASSESSMENT (SCIENCE)
    - UNIT 1 Safety Analysis
    - UNIT 2 Chemistry and Chemical Engineering
    - UNIT 3 Radiological Protection
    - UNIT 4 Control and Instrumentation
  - BRANCH C ASSESSMENT (ENGINEERING)
    - UNIT 1 Structural Integrity
    - UNIT 2 Civil Structures and External Hazards
    - UNIT 3 Mechanical and Electrical Systems, Internal Hazards
    - UNIT 4 Project Management
  - BRANCH D INSPECTION (CHEMICAL PLANT SMALL INSTALLATIONS AND NON-LICENSED SITES)
    - UNIT 1 Inspection: BNF Sites, Drigg
    - UNIT 2 UKAEA and AWE Sites
    - UNIT 3 Project Inspection and Management
    - UNIT 4 Other Sites: Management of Safety, Human Factors
  - BRANCH E INSPECTION (POWER STATIONS)
    - UNIT 1 Inspection Strategy
    - UNIT 2 Site Inspection
    - UNIT 3 Site Inspection
    - UNIT 4 Site Inspection
  - BRANCH F RADIATION AND TECHNICAL
    - UNIT 1 Radiation Protection
    - UNIT 2 Licensing, Emergency Arrangements and Incidents
    - UNIT 3 Nuclear Safety Research

Monday, October 16th, 1995

Plenary Session (3)

Outline of Safety Research of NUCEF

Cochairs

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SAFETY RESEARCH PROGRAM OF NUCEF

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To contribute the safety and establishment of advanced technologies in the area of nuclear fuel cycle, Japan Atomic Energy Research Institute (JAERI) has constructed a new research facility NUCEF (Nuclear Fuel Cycle Safety Engineering Research Facility) as the center for the research and development, particularly on the reprocessing technology and transuranium(TRU) waste management.

NUCEF consist of three buildings, administration building, experiment building A and B. Building A has two experiment facilities STACY(Static Experiment Critical Facility) and TRACY (Transient Experiment Critical Facility). The experiment building B is referred to as BECKY(Back−end Fuel Cycle Key Elements Research Facility). Researches on the reprocessing and the waste management are carried out with spent fuels, high−level liquid waste, TRU etc. in the $\alpha\ \gamma$ cell and glove boxes(see Fig.1).

NUCEF was constructed with the following aims.

Using STACY and TRACY, are aimed

(1) research on advanced technology for criticality safety control,

(2) reconfirmation of criticality safety margin of the Rokkasho reprocessing plant.

Using BECKY, are aimed,

(1) research on advanced technology of reprocessing process,

(2) contribution to develop the scenario for TRU waste disposal,

(3) development of new technology for TRU partitioning and volume reduction of radioactive waste.

To realize the above aims, following 5 research subjects are settled in NUCEF,

(1) Criticality safety research,

(2) Research on safety and advanced technology of fuel reprocessing,

(3) Research on TRU waste management,

(4) Fundamental research on TRU chemistry,

(5) Key technology development for TRU processing.

Criticality Safety Research

Criticality experiment facilities in NUCEF: STACY and TRACY, provide criticality data of solution fuels such as nitrate solutions of low−enriched uranium, plutonium, and their mixture.

Research items using STACY are,

- Expansion of knowledge on criticality safety,
- Revision of criticality safety handbook,
- Realization of rational criticality safety design and control,
- Verification of criticality calculation code and nuclear data,
- Demonstration of criticality safety margin in the design of process equipment.

In STACY, criticality is attained by increasing solution fuel level. Using this facility, are measured basic criticality data such as critical volume temperature coefficient of reactivity and kinetic parameters as well as effectiveness of neutron absorber or neutron interaction effect. For the above experiments, several kinds of core tanks are prepared as shown in Fig.2.

Research items using TRACY are,

- Study on nuclear thermal fluid behavior in the reactor core,
- Improvement of the accuracy in a site evaluation,
- Development of estimation method of exposed radiation dose,
- Demonstration of confinement capability of radioactive materials.

In TRACY, reactivity is added by two ways, feed of solution fuel and control rod withdrawal. Major specifications of TRACY are shown in Table−1. Integral power limit is 32MW・sec($1x10^{18}$fissions).

Criticality safety research in chemical process is performed for,

- Verifying the phenomena under abnormal conditions and studying of "the third phase" to clarify its formation mechanism, etc.,
- Development advanced technology, i.e., application of organic neutron absorber to chemical process of nuclear fuel,
- Development of high precision inline monitor technique for measurement of nuclear fuel solution,
- Experiments to obtain the requisite data for criticality safety design and analysis.

For development of advanced technology, organic neutron absorber, m−carboran, is searched whose structure is shown in Fig.3. The features of it are soluble in organic phase, physically and chemically stable, and little influence on extraction performance. In Fig.4.

reactivity change by the concentration of organic boron compound is shown. Infinite neutron multiplication factor, K_∞ of solvent (30% TBP−dodecane) containing 30g/l plutonium decreases from 1.7 to 0.5 by adding 10g/l of organic boron compound. K_∞ being 0.5 means that the solvent never attain critical instead of its volume or geometry. So that large scale of extraction process can be constructed without concerning nuclear criticality.

Research on Safety and Advanced Technology of Fuel Reprocessing

Fundamental behaviors of radioactive nuclides are studied focusing the long−lived nuclides in particular. Assessment of radioactivity confinement capability under the nominal PUREX condition is conducted to verify the source term which has been assumed for the environmental safety assessment of reprocessing facility.

A new process PARC(Partitioning Conundrum Key Process) is developed aiming at the further reduction of waste generation and radioactivity releases in the future reprocessing. Separation efficiencies are enhanced for I−129 and C−14 in head−end, and Np−237 and Tc−99 in extraction. Am and Cm are recovered from highly radioactive raffinate. Figure 5 shows schematic flow diagram of the PARC process. As shown in this figure, Np and Tc are separated at extraction process.

Research on TRU Waste Management

TRU waste arising from reprocessing and MOX fabrication are treated and disposed into the geological matrix. New types of ceramics are developed as TRU waste forms. Migration behavior of TRU elements is studied in various substances composing engineered and natural barriers. Non−destructive measurement technique using active and passive neutron assay and quality inspection method by CT equipment are developed from the view point of the safety management of TRU waste.

Fundamental Research on TRU Chemistry

Fundamental data on elements are collected and evaluated by conducting characterization of high burn−up spent fuels and basic study on reprocessing based on new principle.

Key Technology development for TRU processing

Key technologies of TRU processing are developed focusing on TRU waste treatment method and process operation system, using the fuel treatment facility for critical experiment in NUCEF. The technologies are applied to enhance the safety operation of reprocessing facility.

Research Collaboration

Since NUCEF is a large research complex for the studies on safety and development of basic and advanced technologies of fuel cycle back−end with a wide variety, it is important to maximize the effectiveness of research activities in the facility by conducting cooperative research with other organizations. For examples, JAERI and CEA of France signed a General Cooperation Agreement, and JAERI and IPSN are cooperating under the general agreement. As shown in Fig.6, under the JAERI−IPSN cooperation agreement, research on criticality safety is carried on. JAERI and CEA are cooperating in the field of fuel cycle radioactive waste and fuel management. NUCEF is expected to play important roles for research collaboration and aims to be new research center as shown in Fig.7.

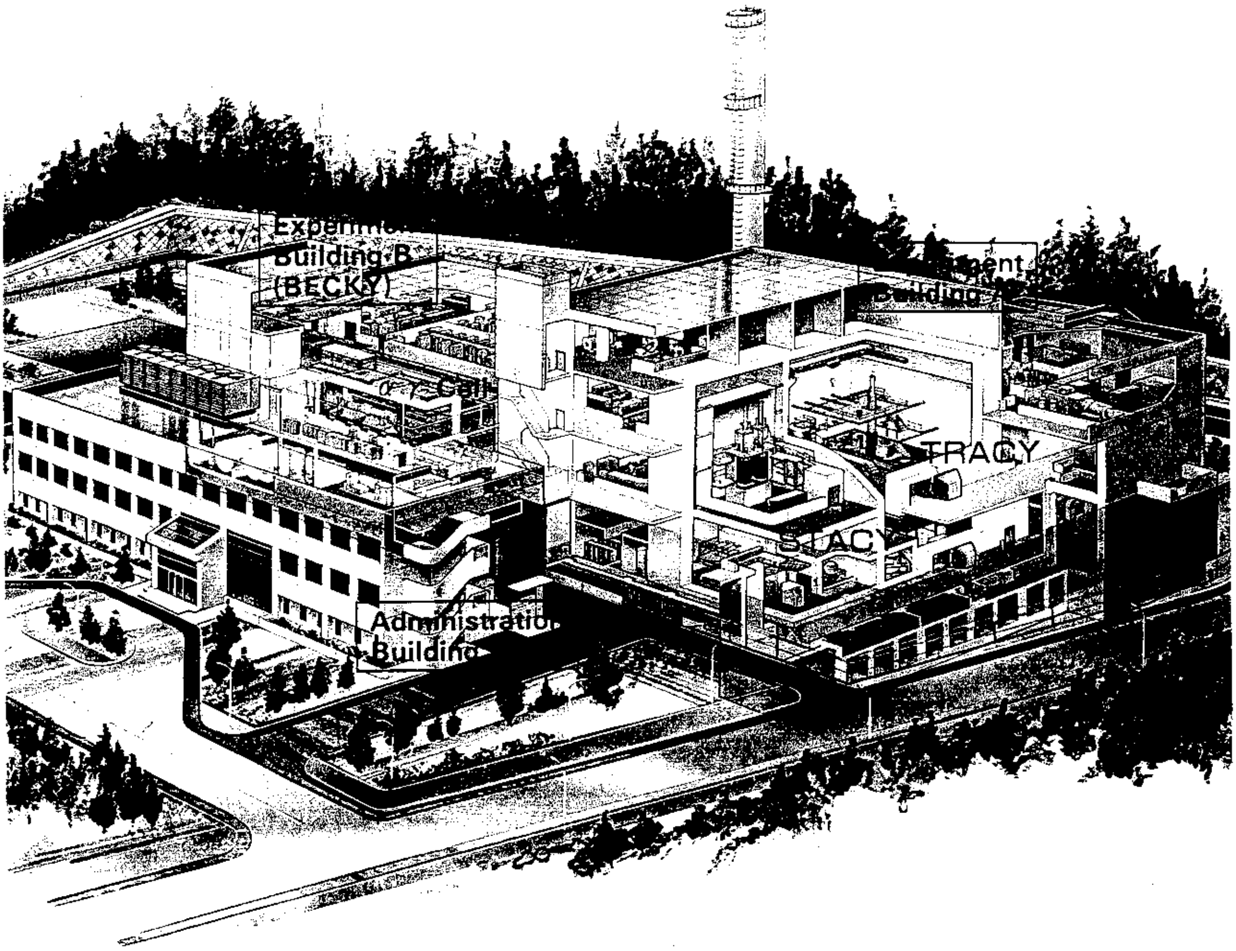


Fig.1 ▲ NUCEF buildings
NUCEF is composed of Administration Building, Experiment Building A and B.

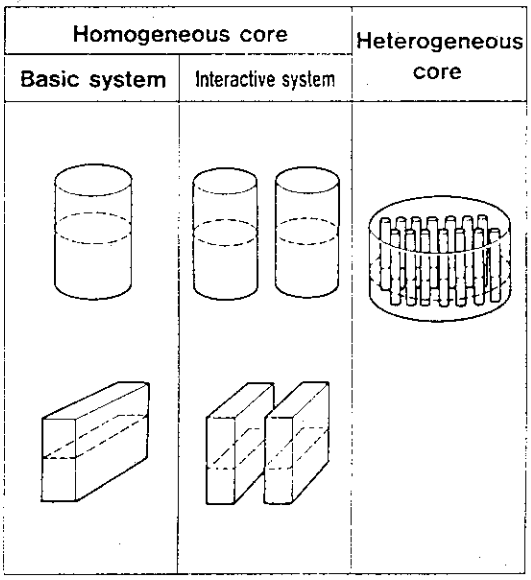


Fig.2 Shape of STACY core tank

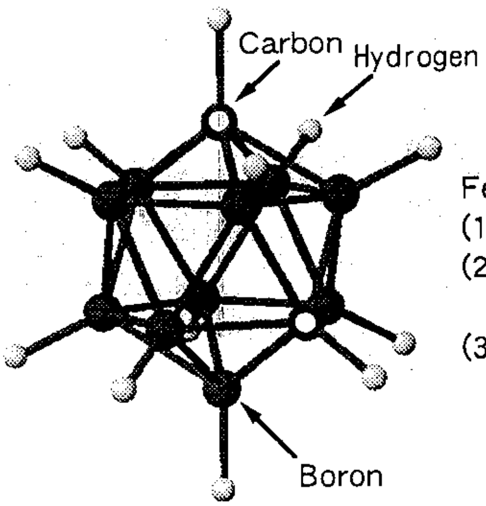
Table.1

TRACY

Transient Experiment Critical Facility

Power	Max. 10kW (stationary operation) Max. 5000MW (transient operation)
Excess Reactivity	Max. 0.8\$ (stationary operation) Type T50 Max. 3\$ Type T80 Max. 2\$ (transient operation)
Core volume	Type T50 Max. $0.2m^3$ Type T80 Max. $0.5m^3$
Reactivity control method	Feed and drainage of solution fuel Withdrawal of a transient rod

Major specification of TRACY



Features;
(1) Soluble in organic phase,
(2) Physically and chemically stable, and
(3) Little influence on extraction performance

Structure of m-carborane ($H_{12}B_{10}C_2$)

Fig.3 Criticality Safety Design by Using Soluble Neutron Poison

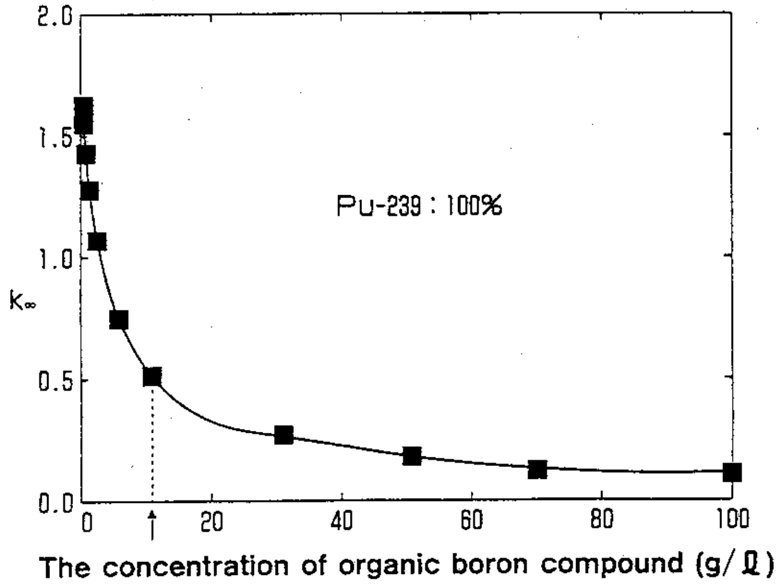


Fig.4 Effect of organic neutron absorber

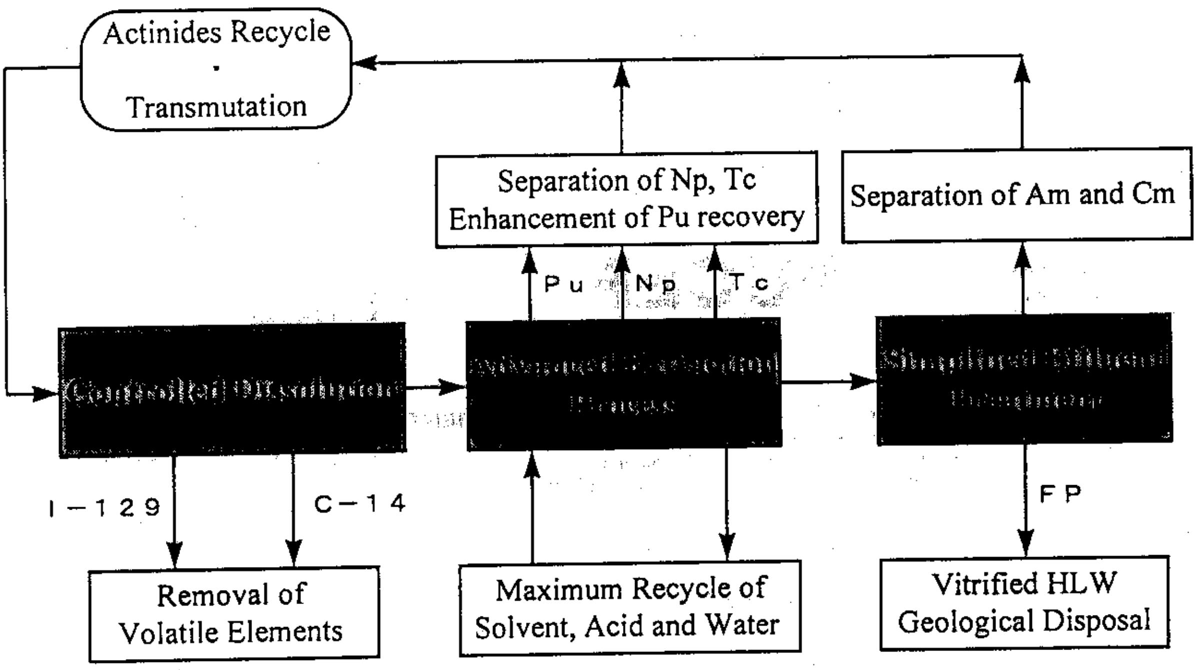
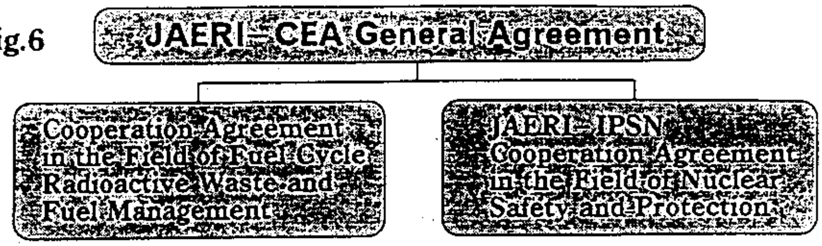


Fig.5 Advanced Reprocessing Incorporated with Partitioning Function (PARC PROCESS)

Fig.6



Specific Topic of Cooperation

(1) Research on TRU and long-lived radionuclides partitioning

(2) Migration Behavior of long-lived nuclides in geological media

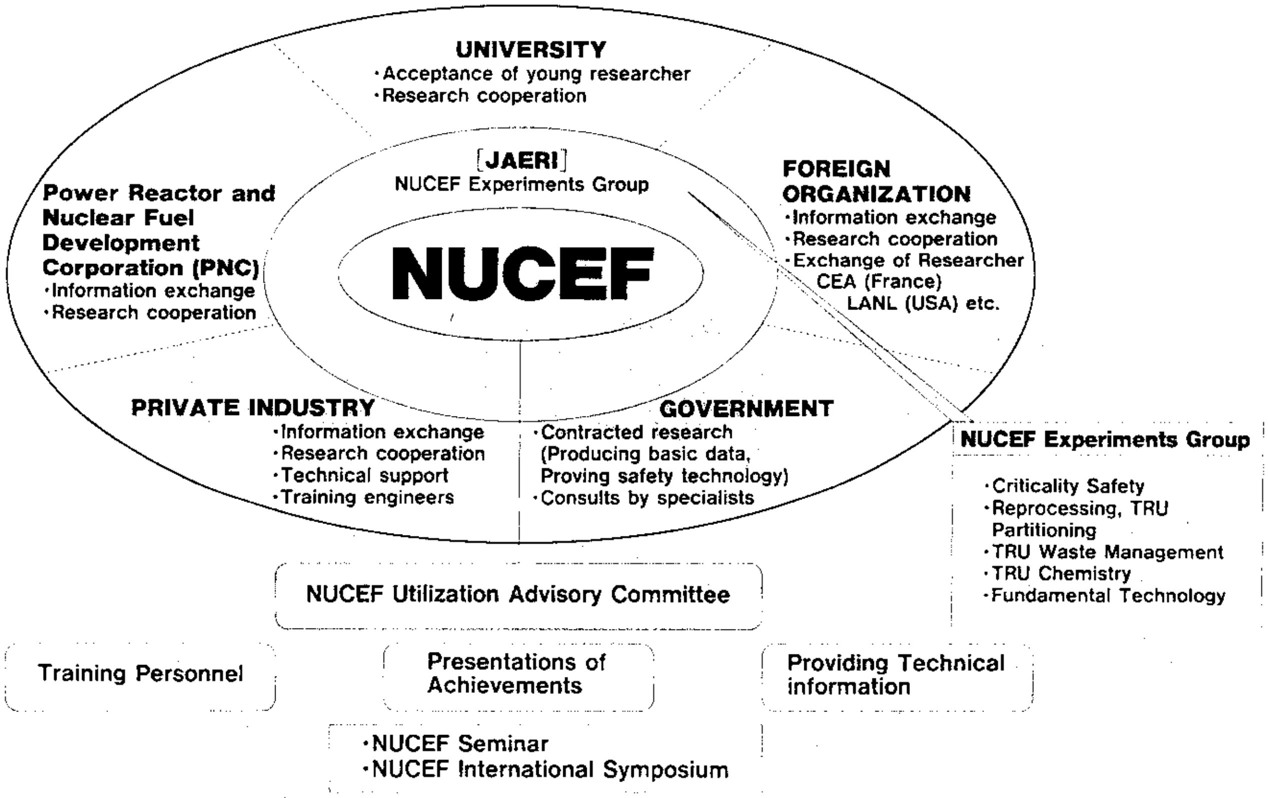
(3) Studies on nondestructive measuring and testing techniques of TRU waste packages

Specific Topic of Cooperation

(1) Criticality Safety

Fig.7 **New Research Center NUCEF**

Research Promotion by Cooperation with Other Organization



Outline of NUCEF Facility

I. TAKESHITA

Department of NUCEF Project
Japan Atomic Energy Research Institute

Introduction

NUCEF is a multipurpose research facility in the field of safety and advanced technology of nuclear fuel cycle back-end. Various experiment facilities and its supporting installations, in which nuclear fuel materials, radio isotopes and TRU elements can be handled, are arranged in more than one hundred rooms of two experiment buildings. Its construction was completed in middle of 1994 and hot experiments have been started since then.

Facility Layout

NUCEF is located on the site (30,000m^2) of southeastern part in the Tokai Research Establishment of JAERI facing to the Pacific Ocean. The base of Experiment Buildings A and B was directly founded on the rock existing at 10~15m below ground level taking the aseismatic design into consideration. Each building is almost same sized and composed of one basement and three floors of which area is 17,500m^2 in total (Fig. 1). In the basement, there are exhaust facilities of ventilation system, treatment system of solution fuel and radioactive waste solution and storage tanks of them. Major experiment facilities are located on the first or the second floors in each building. An air-inlet facility of ventilation system for each building is equipped on the third floor.

Most of experiment facilities for criticality safety research including two critical facilities: Static Experiment Critical Facility (STACY) and Transient Experiment Critical Facility (TRACY) are installed in Experiment Building A. Experiment equipments for research on advanced fuel reprocessing process and on TRU waste management, which are named BECKY (Back End Fuel Cycle Key Elements Research Facility), are installed in laboratories and a-g cells in Experiment Building B (Table 1, Fig. 2).

Construction History of NUCEF

Following the arrangements of research programs and the conceptual and basic design of the facility, internal safety review was initiated on 1986. On October 1988, construction license was given by the government after the safety review for more than one year. It took six years to complete the construction of NUCEF. After 150kg low enriched uranium dioxide was dissolved into nitrate solution, the first criticality test was successfully achieved on February 23 this year. Hot experiments in BECKY were also started from the beginning of this year. The first criticality test of TRACY is planned on coming December (Table 2).

Facilities for Criticality Safety Experiments

Major facilities for criticality safety research in NUCEF are STACY, TRACY and Fuel Treatment System.

(1) STACY

STACY was designed so as to obtain critical mass data of low enriched uranium and plutonium nitrate solution which are extensively handled in an LWR fuel reprocessing plant. The core tank of STACY is replaceable to obtain critical mass data of different geometry and size of tanks. Moreover, not only single core system but interacting system with two core tanks can be formed. The core system is installed inside a reactor hood in a reactor room because of the contamination control. Major specification of STACY is listed in Table 3. Reactivity control of STACY is made by feed and drainage of solution fuel to/from the core (Fig. 3).

A contact needle type level meter is used to measure the solution level in the core. Two additional needles are attached to the tip of the level meter to detect the over feeding of fuel solution and to trigger off reactor shutdown (Fig. 4).

(2) TRACY

TRACY is a critical facility by which critical burst is demonstrated with low enriched uranium nitrate solution. Major specification is listed in Table 4. Maximum integrated power is limited to 32MW•sec (1×10^{18} fission). On steady state operation of TRACY, reactivity control is made by feed and drainage of solution fuel to/from the core tank like STACY. For the transient operation, two methods can be applied to add reactivity to the core. One is withdrawal of a transient rod inserted in the center of the core by pressurized air or by an electric motor. The other is feeding of solution fuel to the core tank beyond the critical level (Fig. 5). Another difference from STACY is the vent line system which is directly connected to plenum of core tank and forms a closed loop. Along the vent line, sampling points of aerosol released from solution fuel and measurement devices of iodine are equipped to investigate behavior of aerosol including fission products during critical excursion. This closed vent line system has also the safety function to dilute hydrogen gas and to reduce radio activity that are generated in the core during the experiment (Fig. 6).

(3) Fuel Treatment System and other subsidiary equipments

This system is designed exclusively to arrange solution fuel for STACY and TRACY experiments, which enables the experiments with wide range of composition and concentration of nuclear fuels in both critical facilities. Dissolution, dilution and concentration, separation and solution storage are main functions of the system (Fig. 7). These functions resemble those of reprocessing process, however its capacity is just laboratory scale and all equipments are installed in glove boxes because of low radio activities of solution fuel.

Various destructive and non-destructive analysis equipments are installed in Analytical Laboratories in Experiment Building B to support the critical experiment and the operation of fuel treatment system as well as the nuclear material accountancy (Fig. 8). Some of those analytical equipments, which are listed in Table 5, are used for experiments in BECKY. Among them, Hybrid K-edge/XRF densitometer has been developed and fabricated by Los Alamos National Laboratory under the US DOE/JAERI research cooperation (Fig. 9).

In Alpha-chemical laboratory in Experiment Building A, basic experiments are carried out on process anomaly which might potentially cause critical accidents. In the same laboratory, an in-line test loop is installed for the development of in-line monitor for reprocessing process (Fig. 10).

Facilities in BECKY

Experiment facilities and equipments in BECKY are listed in Table 1. Major experiment equipments are installed in hot cells and 27 glove boxes in several laboratories in Experiment Building B.

(1) Hot Cells

The hot cells consist of three cells; a process cell, a chemical cell and a loading cell. In the process cell, laboratory-scale experiment equipments are installed for research on reprocessing and partitioning processes called PARC Process and Four-group Partitioning of high level liquid waste (HLLW), respectively (Fig. 11, Fig. 12). Spent fuel specimens up to 45,000MWD/t (3kg/year) and actual HLLW (5,000Ci/year) can be handled in the process cell for both experiments. Small amount of spent fuel up to 72,000MWD/t can be treated for research on TRU chemistry in the chemical cell adjacent to the process cell.

(2) Analytical Experiments in Glove boxes and Hoods for Reprocessing Experiments

Some of the glove boxes are attached to the hot cells for iodine treatment research, analysis of dissolution off-gas, and sampling of test specimen (Fig. 13). Chemical analysis with small amount of TRU or radio isotopes are conducted in glove boxes and hoods which are installed in three laboratories of Experiment Building B.

(3) Experiment Equipments for TRU Waste Management Research

Experiment equipments for ceramic solidification of TRU waste and for measuring migration behavior of TRU elements in natural and engineered barriers are installed in nine glove boxes in a TRU waste laboratory (Fig. 14, Fig. 15). Originally designed non-destructive measurement systems for TRU waste forms are installed in a laboratory in the basement of Experiment building B. This measurement systems are composed of experiment equipments with passive and active assay using neutrons and computed tomography equipment using gamma-ray (Fig. 16).

Provisions for Future research program

There are a few rooms remained for future use in NUCEF. One is a space in a laboratory II whose floor sustains heavy steel cells (Fig. 17). Another one is a room adjacent to reactor room T where was originally a space of tall pulse columns tests for criticality safety research. Preparation of ventilation is already made to both rooms. Some of the rooms will be also supplied for future use if their research program is finished.

Current schedule of NUCEF

Hot experiments has just been initiated in NUCEF. Critical experiments of STACY with plutonium solution is planned when all equipments for plutonium handling are completed. Criticality burst experiment with TRACY is to be started after hot performance tests by the regulatory body is finished in middle of next year. However, some basic data on criticality excursion with low enriched uranium nitrate solution are expected to be obtained even in the performance tests.

BECKY will gradually enter into the phase of hot experiments with irradiated fuel or actual HLLW, and steel cell installation is expected to be started within a few years (Fig. 18).

Table 1 Facility Groups in NUCEF

Research Area	Major Facilities	Regulation
Criticality Safety	- STACY - TRACY - Fuel Treatment Facility - Chemical Analysis Equipments	Reactor, RI
Reprocessing (Group Partitioning)	- $\alpha\,\gamma$ cells - PARC Process Experiment Facility - Four Groups Partitioning Facility - Experiment Equipments for TRU chemistry	Nuclear Fuel Facility (BECKY), RI
TRU waste	- Equipments for TRU waste Solidification - Performance Test Equipments for Barriers - Equipments for Passive and Active Assay	

Table 2 Construction Milestones

1986 ~	**Safety Assessment**
1988 Oct.	**Construction License**
1993	**Completion of Buildings**
1994	**Function Test in Cold State**
1994 Sep.	**Dissolution of Uranium Dioxide**
1995 Feb.	**First Criticality of the STACY**
1995 Dec.	**First Criticality of the TRACY**

Table 3 Major Specification of STACY

Power	Max. 200W
Excess Reactivity	Max. 0.8$
Fuel	U nitrate solutionm, Pu nitrate solution and Mixture nitrate solution
Isotope ratio	^{235}U enrichment: 4%, 6% and 10% ^{240}Pu ratio: 5%~25%
Core Configuration	Basic homogeneous core: Unit cylindrical or slab tank Interacting homogeneous core: Identical cylindrical or slab tanks Heterogeneous core: A cylindrical tank and fuel rods
Core Dimension	Height of fuel solution part: 40cm~140cm Radius of cylindrical core: 21cm~100cm Thickness of slab core: 10cm~50cm Width of slab core: 70cm(Fixed)
Reactivity Control Method	Feed and drainage of fuel solution
Shut Down Method	Normal operation: Drainage of fuel solution Emergency: Insertion of safety rods or sheets

Table 4 Major Specification of TRACY

Power	Static operation mode: Max. 10 kW Transient operation mode: Max. 5GW
Integrated Power	Max. 32 MW-sec (1×10^{18} fissions)
Fuel	Uranium nitrate solution Enrichment: 10% Concentration: Max. 500 gU/lit.
Reflector	none or water
Core Dimension	Shape: Annular Inner diameter: 10cm Outer diameter: 50cm or 80cm Height of fuel solution part: 40cm~100cm (Height of core tank is 200cm)
Initial Temperature	<40 °C
Maximum Pressure	880 kPa

Table 5 Major Analytical Equipment

Equipment	Analytical Method	Application
Density Meter	Vibration Tube Method	Solution Density
Mass Spectrometer	Surface Ionization Mass Spectrometry	Isotope Composition
Automated Potentiometric Titrator	Redox Titration Neutralization Analysis	Concentration of U or Pu Concentration of HNO_3 or TBP
Hybrid K-edge/XRF Densitometer	K-edge Absorption X-ray Fluorescence Spectrometry	Concentration of U or Pu
γ-ray Spectrometer	γ-ray spectrometry	Concentration of γ nuclides
α Spectrometer	α Spectrometry	Concentration of α nuclides
ICP Emission Spectrophotometer	ICP Emission Spectrophotometry	Concentration of impurities
UV Spectrophotometer	Absorption Spectrophotometry	Concentration of U, Pu Impurities, etc.
IR Spectrophotometer	Absorption Spectrophotometry	Concentration of TBP residues in solution
Liquid Scintillation Counter	Liquid Scintillation Counting	Concentration of ^{3}H, ^{14}C, etc.
NaI Scintillation Counter	NaI Scintillation Counting	γ Activity
Gas-flow Counter	2π Gas-flow Counting	α and β Activity

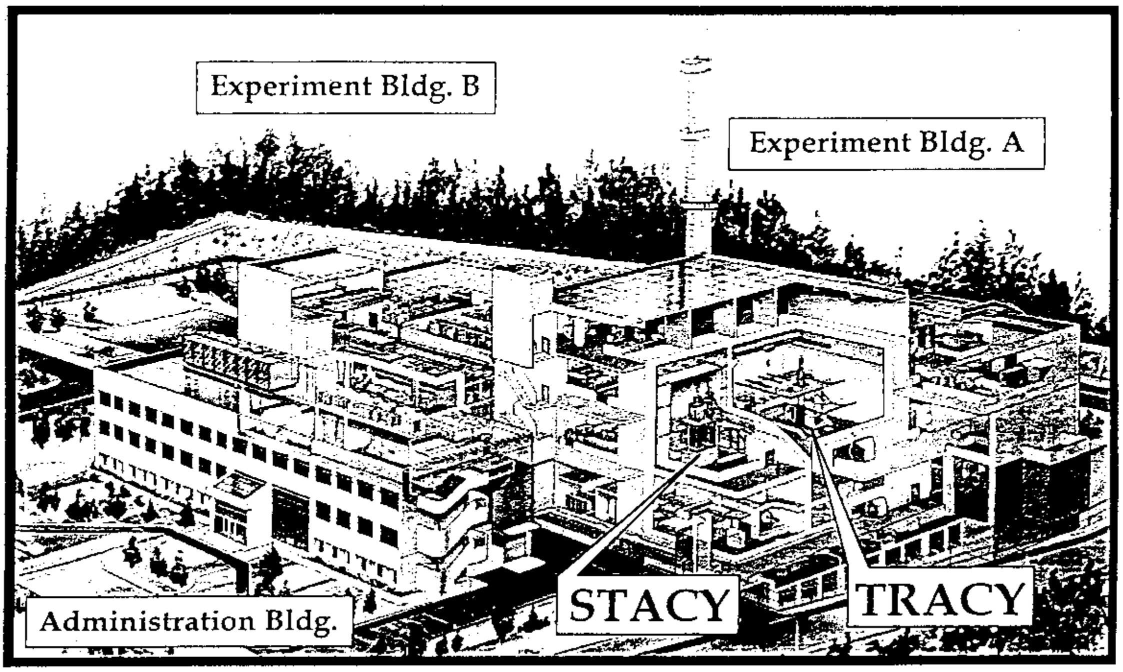


Fig. 1 Bird's eye view of NUCEF

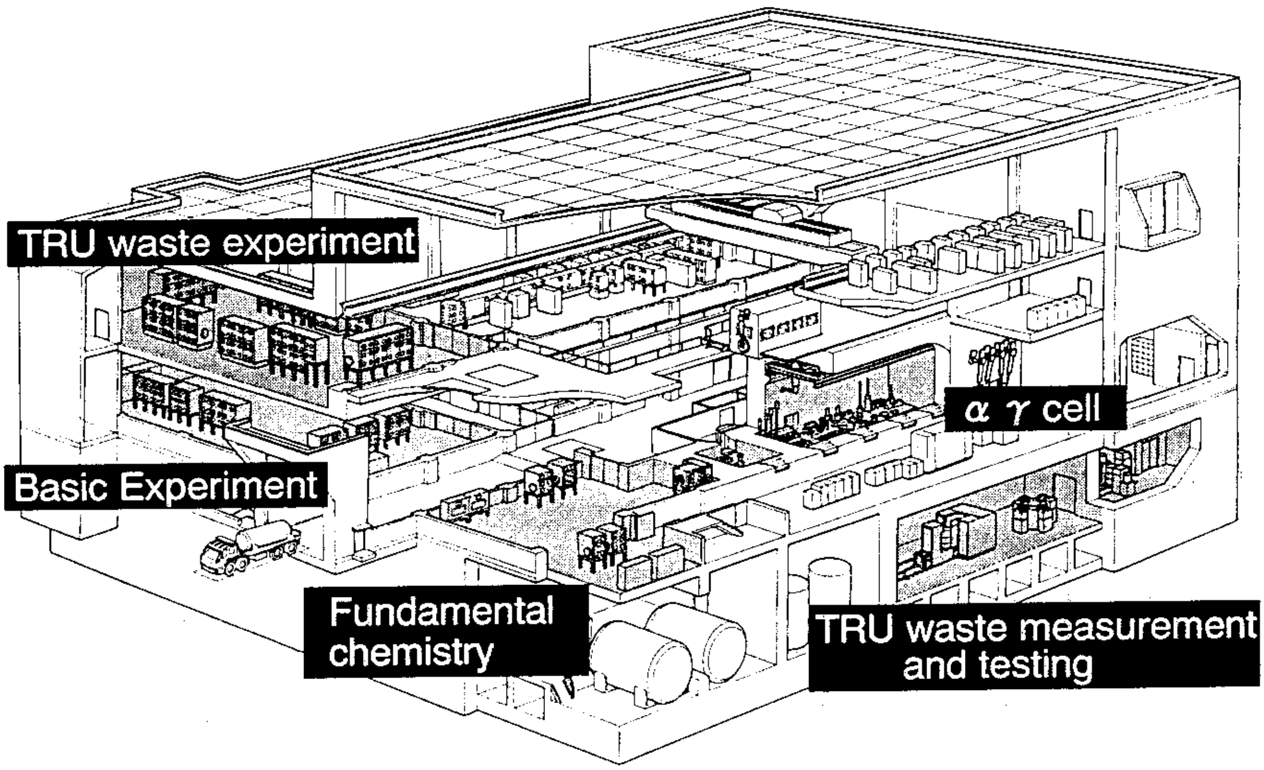
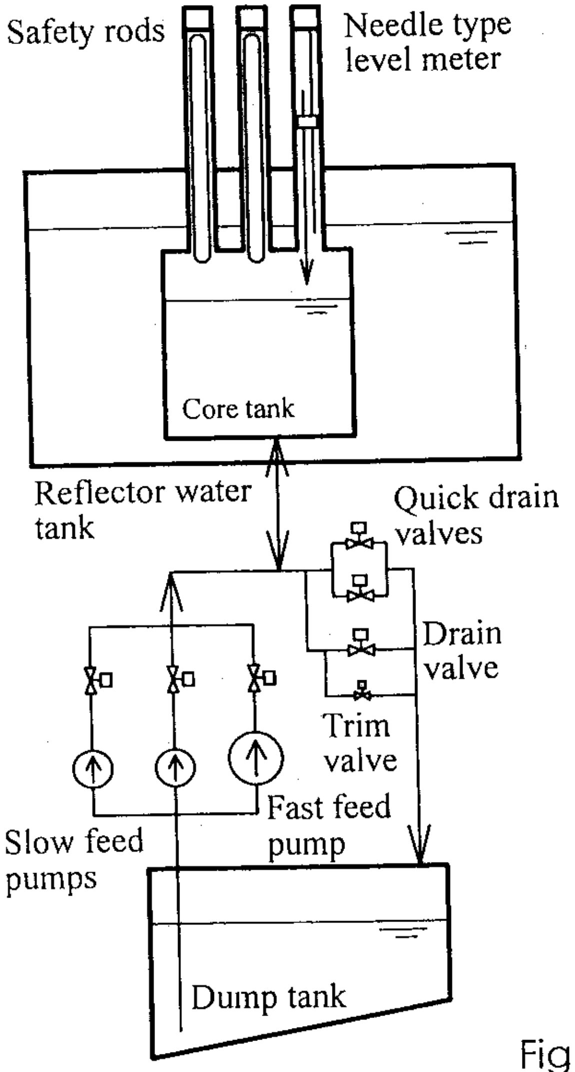


Fig. 2 Bird's eye view of Experiment Building B (BECKY)



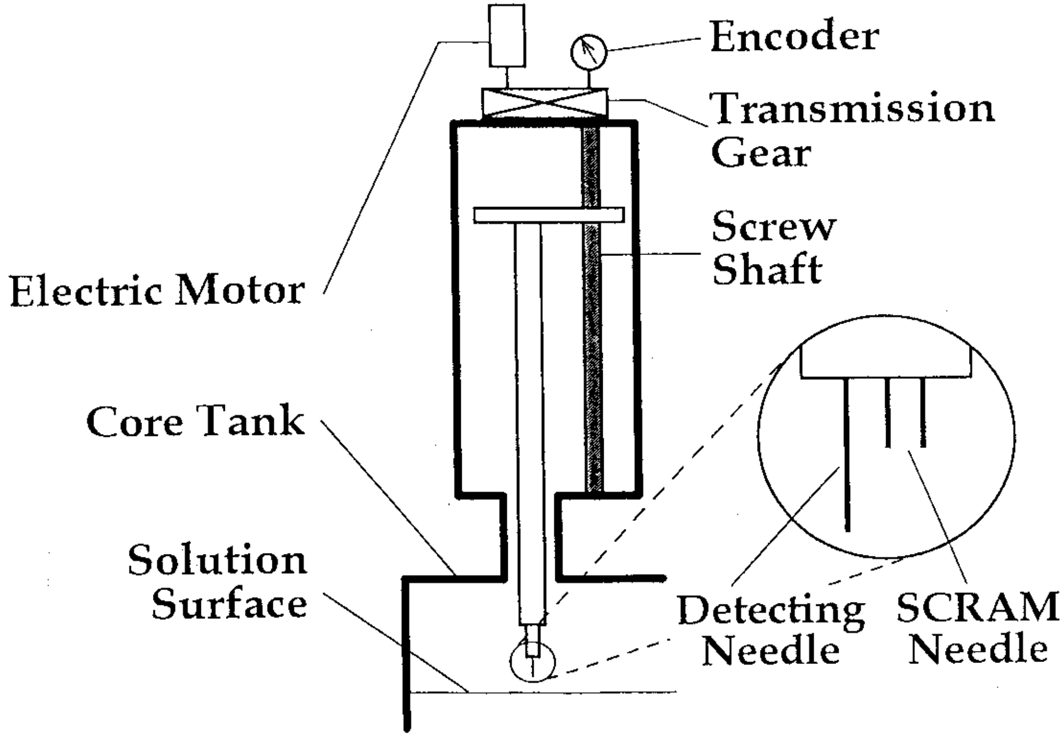


Fig. 4 Level Meter of STACY

Fig. 3 Schematic Flow Diagram of STACY

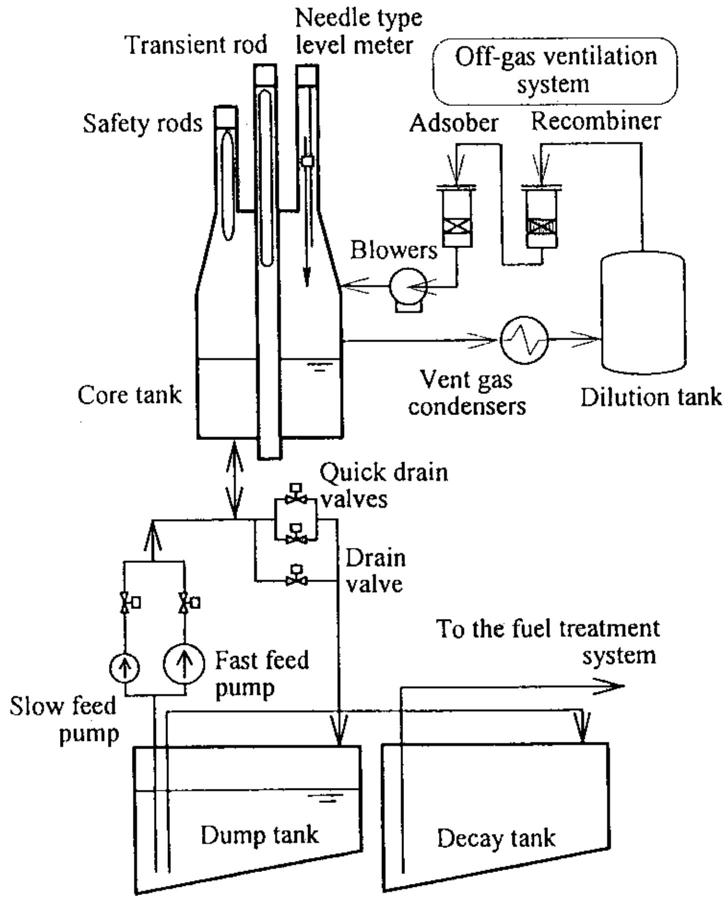


Fig. 5 Schematic Flow Diagram of TRACY

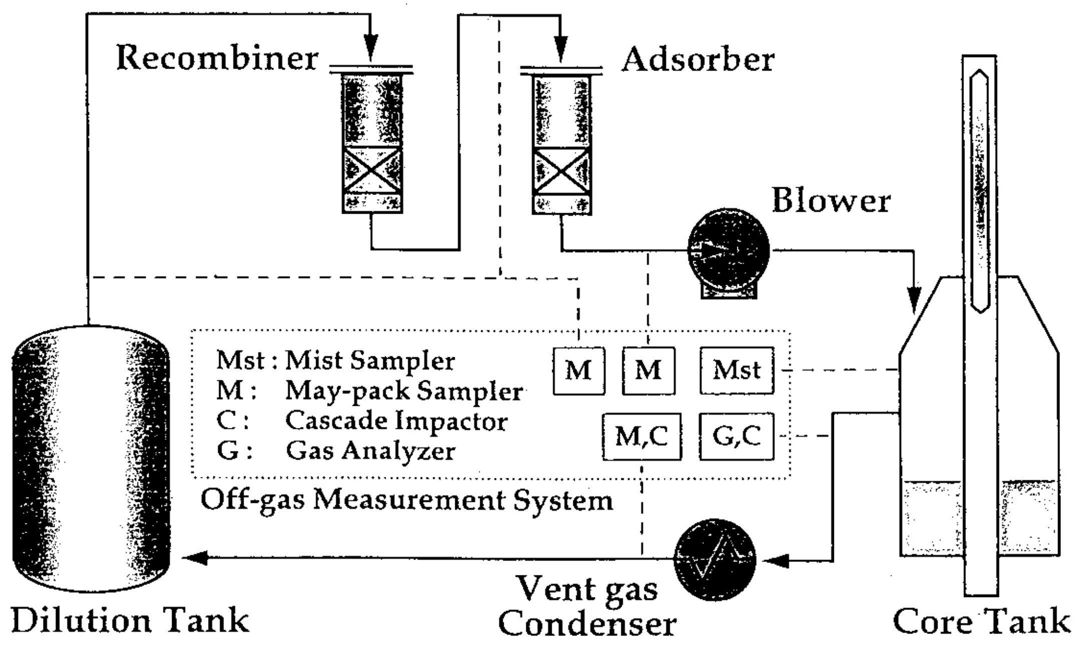


Fig. 6 Off-gas Ventilation System

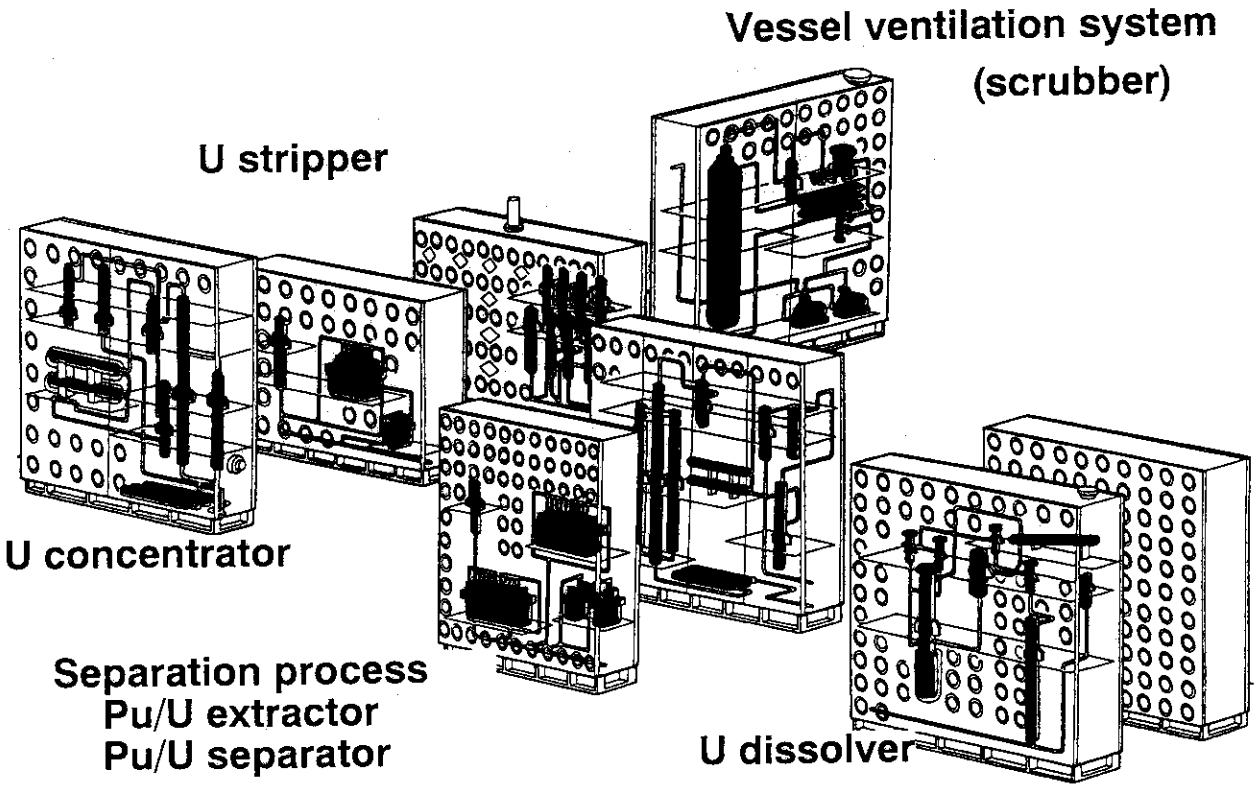


Fig. 7 Layout of Fuel Treatment System

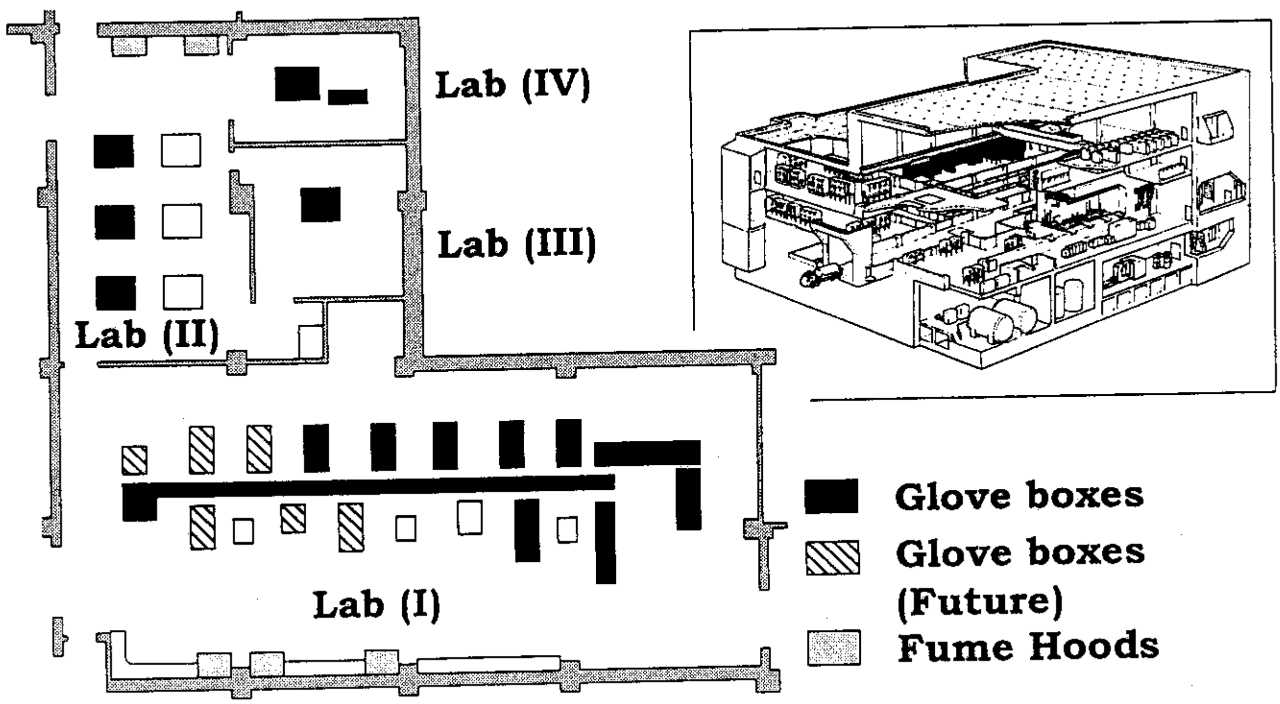


Fig. 8 Layout of Analytical Laboratories and Instruments

Fig. 9 K-edge Absorption/X-ray Fluorescence Spectrometry

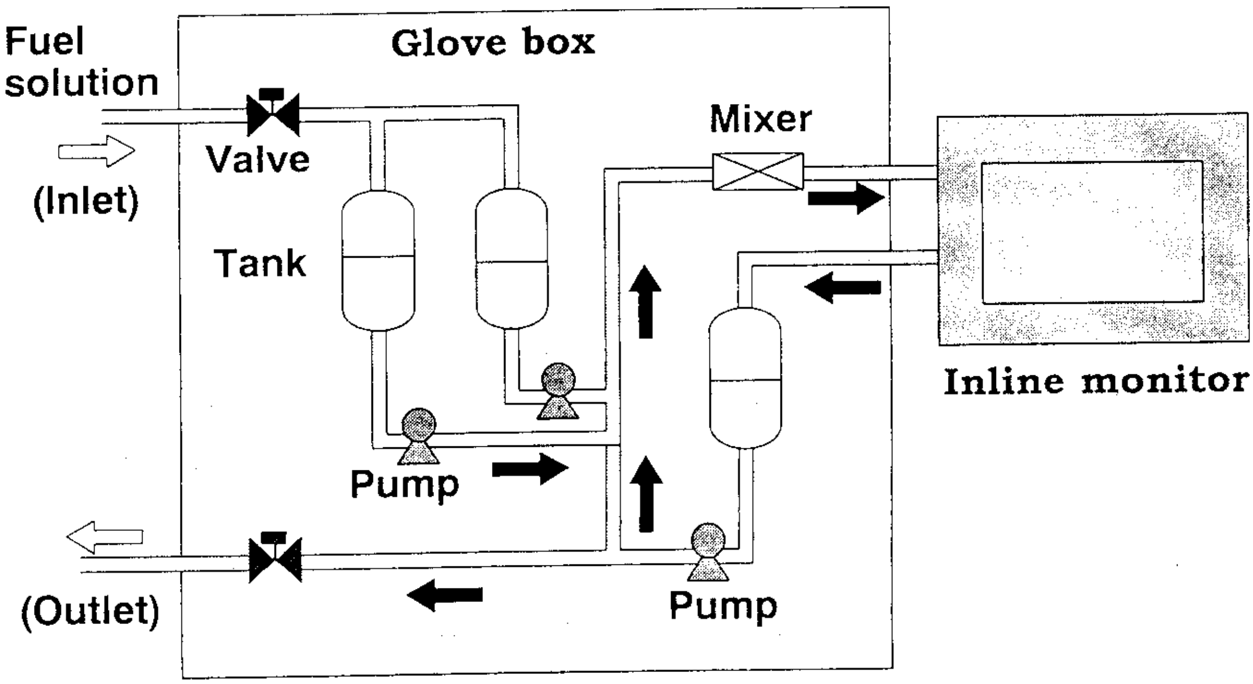


Fig. 10 Schematic Flow Diagram of Equipment for Inline Monitoring Test

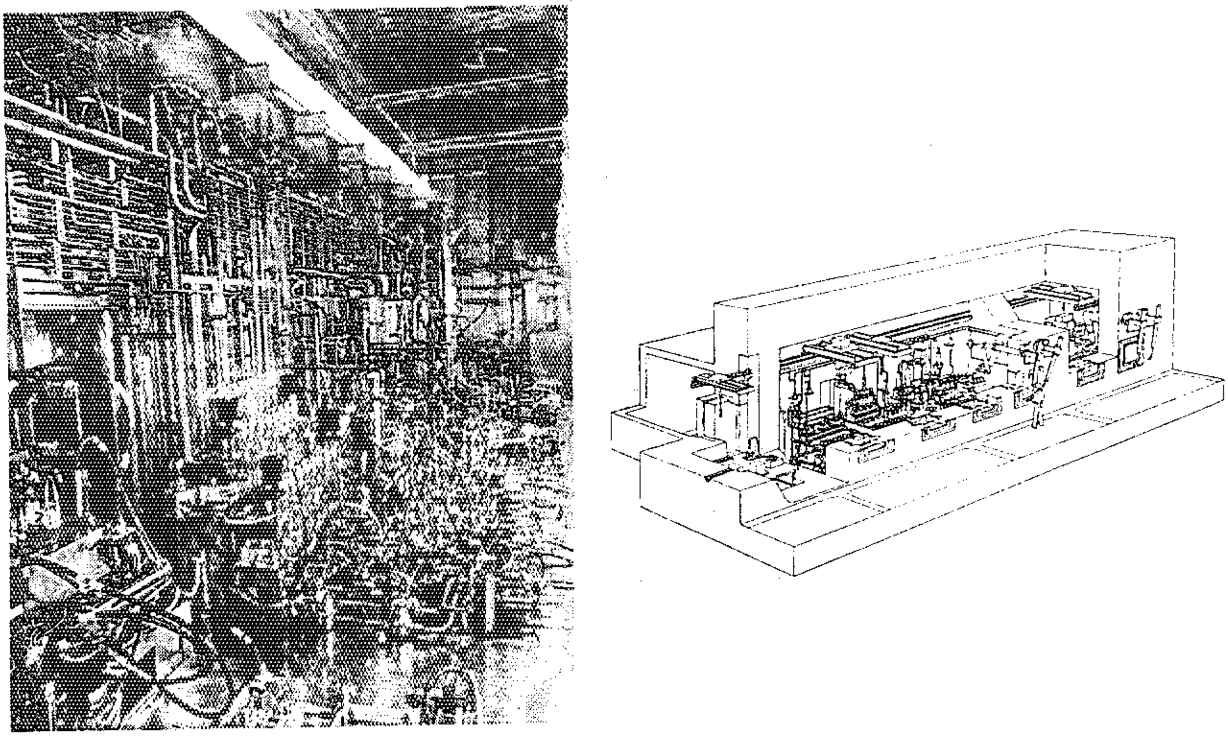


Fig. 11 Reprocessing Process Experiment Equipment

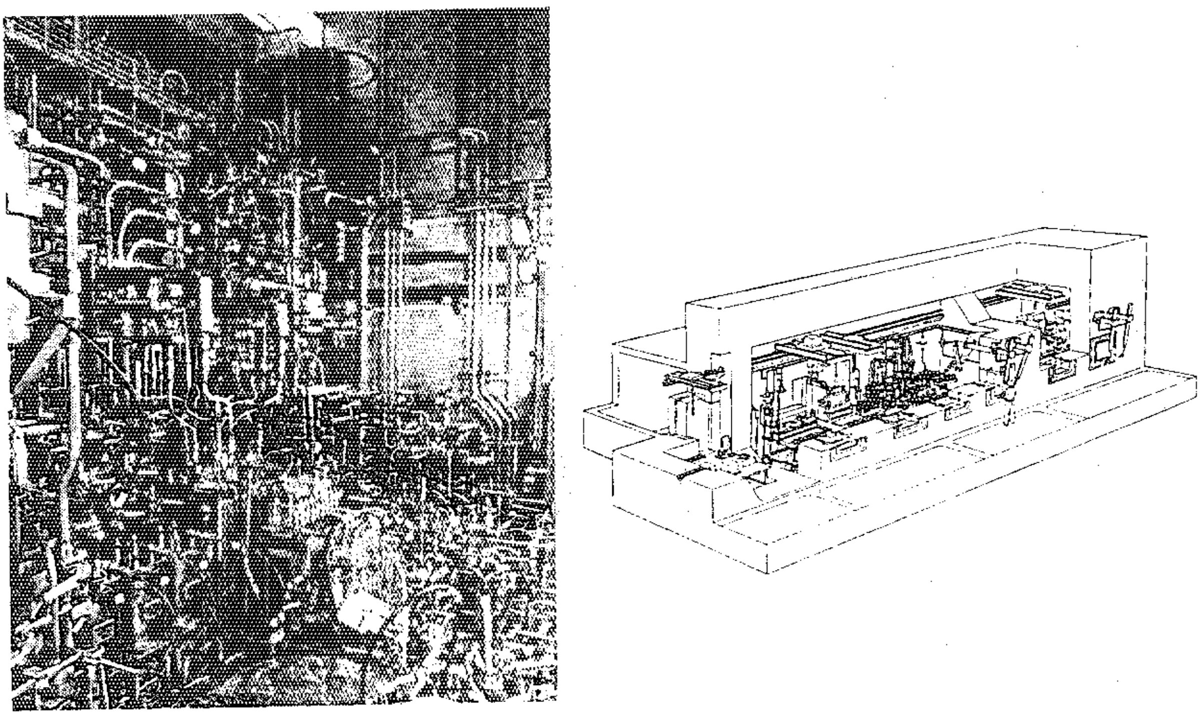


Fig. 12 Equipments of Partitioning Process

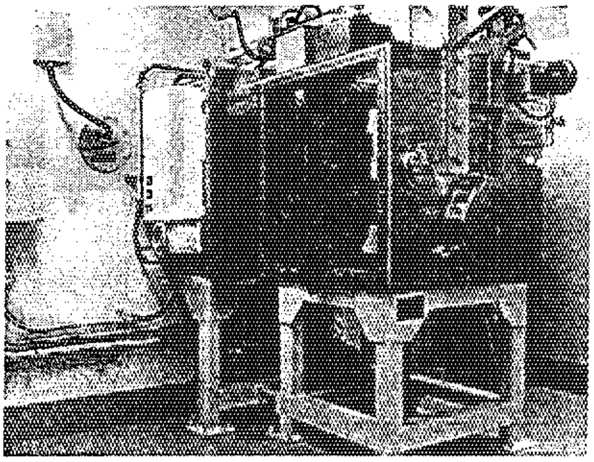


Fig. 13 Sampling Box

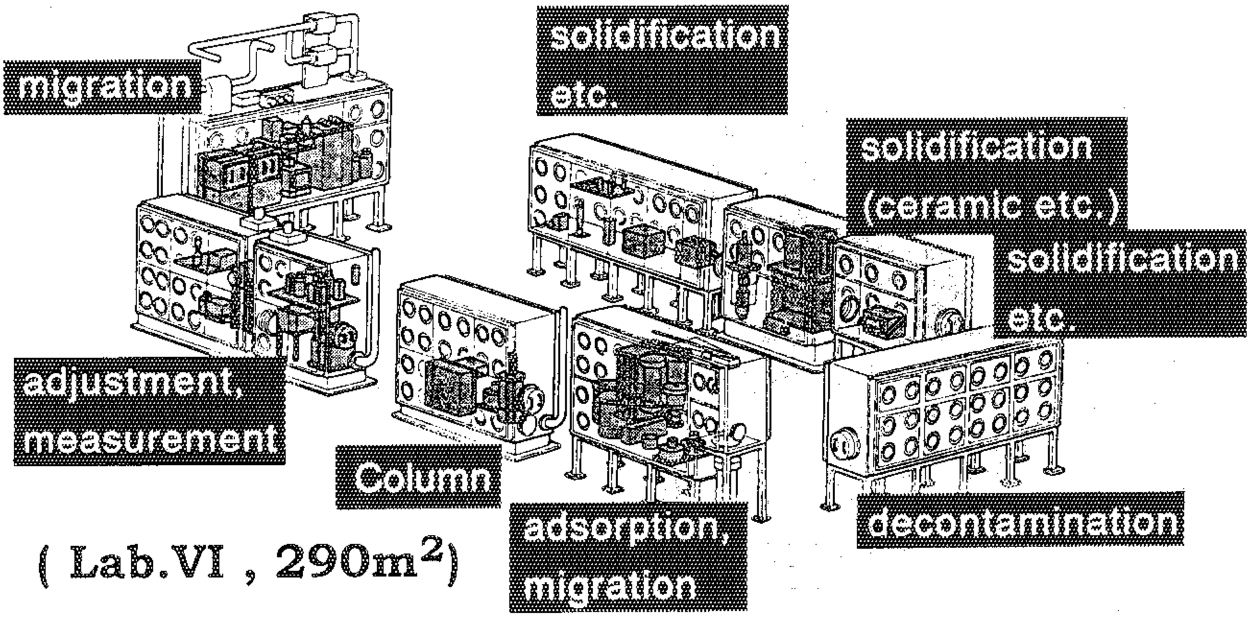


Fig. 14 Glove Boxes for Research on TRU Waste Treatment and Disposal

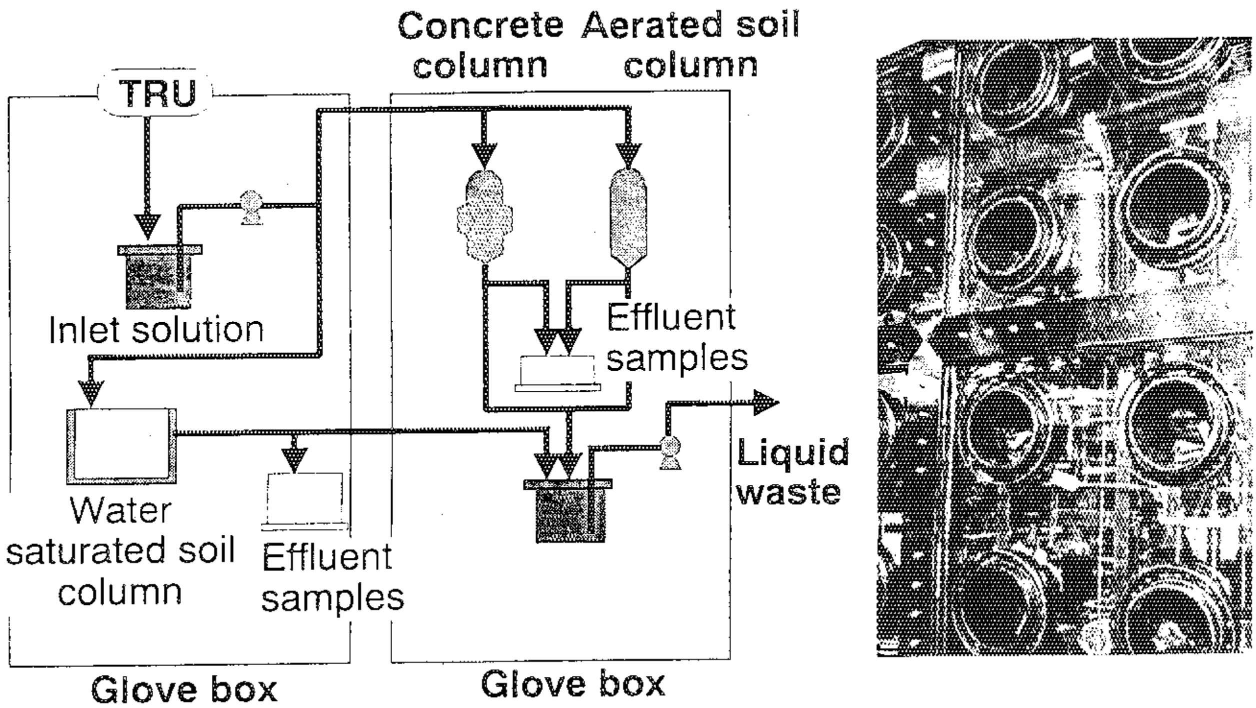
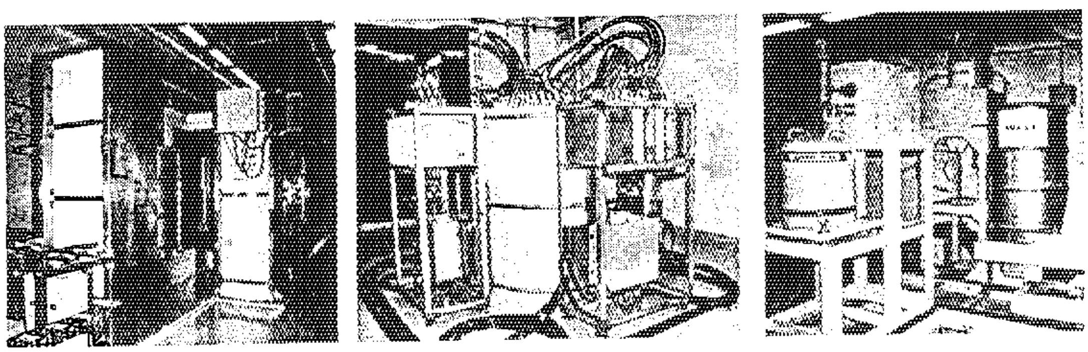


Fig. 15 Performance Test for the Evaluation of Natural and Engineered Barriers



Active assay Passive assay Computed tomography

Fig. 16 Equipment for TRU Waste Measurement

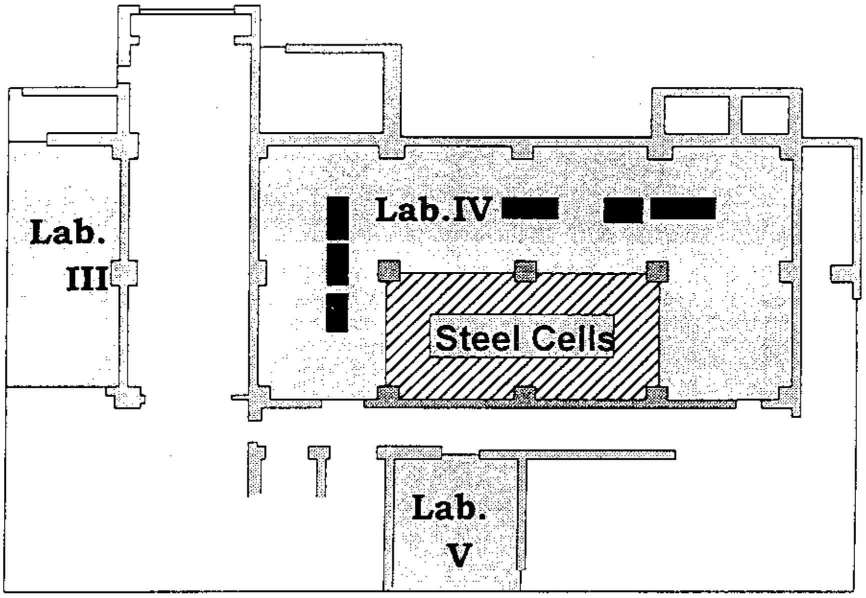


Fig. 17 Key Plan of the Steel Cell Installation

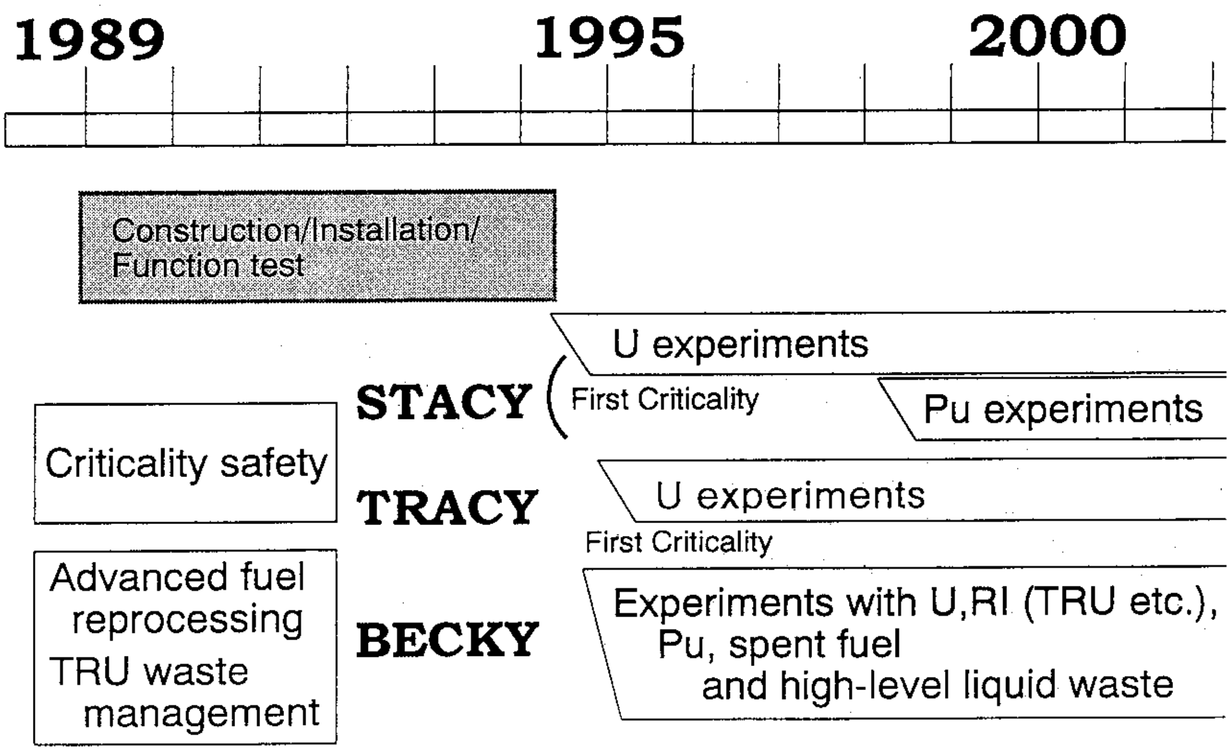


Fig. 18 Schedule of NUCEF Project

Tuesday, October 17th, 1995

Technical Session (1)

Criticality Safety

Cochairs

K.Nishina(Nagoya Univ., Japan)
P.Watson(HSE/NII, UK)

REVIEW OF THE INTERNATIONAL CONFERENCE ON NUCLEAR CRITICALITY—ISSUES, DISCUSSIONS, AND CHALLENGES

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Presented at
NUCEF'95
The First NUCEF International Symposium
October 16-17, 1995
Hitachinaka, Ibaraki, Japan

*Managed by Lockheed Martin Energy Systems, Inc., under contract DE-AC05-84OR21400 with the U.S. Department of Energy.

REVIEW OF THE INTERNATIONAL CONFERENCE ON NUCLEAR CRITICALITY—ISSUES, DISCUSSIONS, AND CHALLENGES

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Oak Ridge, Tennessee, USA

INTRODUCTION

The Fifth International Conference on Nuclear Criticality Safety (ICNC'95) was held September 17-22, 1995, in Albuquerque, New Mexico, U.S.A. Organization and support for the conference was provided by the Sandia National Laboratories (SNL), the Los Alamos National Laboratory (LANL), the University of New Mexico, and the Organization for Economic Cooperation and Development (OECD). This conference traces its history back to 1981 when a group of select criticality safety specialists (mostly experimentalists) from France, Germany, Japan, the United Kingdom, and the United States participated in a small conference at LANL in the United States. The motivation for the conference had been provided by Dr. J. C. Manaranche of France who had asked D. Smith and G. E. Whitesides of the United States if it would be possible for the French experimentalists to be able to visit the experimental facilities at LANL. This first conference was followed by a similar conference held in Dijon, France, in 1983. Then in 1987 the conference was hosted by the Japanese and opened to much wider participation by criticality safety specialists involved in experiments, methods development and analysis, and operations. With the 1987 conference in Japan and the fourth conference (ICNC'91) held in the United Kingdom, the interest and international participation by the criticality safety community has grown rapidly. With this background, the occasion of ICNC'95 was one of much expectation.

THEME AND SPECIAL SESSIONS

The conference focused on the theme, "A Half Century of Nuclear Criticality Safety," with special sessions that provided a historical perspective on the technical accomplishments of the last fifty years. The first of these special sessions was the Pioneers Session held on Sunday evening and featuring speakers from France (P. LeCorche), the United Kingdom (C. Chatburn), Japan (R. Kiyose), and the United States (D. Callihan). These speakers provided a perspective on the origins and directions of criticality safety practice over the last fifty years. The presentations served to highlight the "pillars" of criticality safety: experiment, computation, and practical application.

The pioneers indicated the importance of early experiments to help understand the physics of particle interaction in fissile material systems and to provide data and prototypic demonstration for handling and production processes. Early analyses were simple (in view of today's high-speed computations) processes to aid the experimentalist in the design of a new system or aid in transitioning the narrow gap from the experimental data to the production environment. Analyses have grown increasingly complex as computer programs have evolved to model detailed geometries and to simulate the physics of particle interaction. The criticality safety specialists responsible for the design or operation of a facility, system, or process must use the data obtained from experiment or computation as tools to help ensure a safe environment. Similarly, the experimentalist and the analyst (or code developer) should be mindful of the needs and concerns of the practitioner as experiments are designed, methods and data are revised, and analyses are performed. The Pioneer Session left one with a keen sense that these pillars (or areas) of criticality safety must be appropriately considered, and if any one pillar is relied on too heavily, then safety may be compromised. One definitive concern in this regard was the decline in the number of critical experiments being performed. Another resounding theme from the pioneers was the fear that safety could be compromised under the weight of the additional documentation needed to justify or perform an experiment, defend the validity of a code or data library, and prepare a safety assessment. The concern is that the criticality safety community (and the approving management) will get so involved in completing or overcoming required documentation that important work may be left undone or the understanding of safety concepts may be lost.

The ICNC conference in the United Kingdom in 1991 was the first time that there were participants from the former Soviet Union. This participation was greatly increased at ICNC'95 and was tied into the theme of "A Half Century of Nuclear Criticality Safety" by having a plenary session of Russian speakers who highlighted the history and achievements related to criticality safety in the former Soviet Union. The speakers represented the four major nuclear research facilities in Russia: Kurchatov, Arzamas-16, Obninsk, and Chelyabinsk. The presentations demonstrated the willingness of the Russians to share their data and experiences with the rest of the world. One of the major discussion topics among attendees of ICNC'95 was the availability of an extensive amount of new and expanded experimental information available from Russia that has practical applications to fissile material operations ongoing or planned throughout the world. Of particular interest in the area of experiences was the paper from Obninsk entitled "A Review of Criticality Accidents Which Occurred in the Russian Industry." This paper discussed briefly 12 criticality accidents and their consequences, none of which had been widely available in the literature. The accidents reported in the paper covered a wide range of initiating events and fissile material. More technical details on these accidents will surely benefit the ongoing worldwide work to better simulate accidents and predict their consequences.

GENERAL TECHNICAL PROGRAM

The ICNC'95 conference had 288 attendees from 15 countries. The technical committee reviewed and accepted 145 papers: 59 from the United States, 25 from the United Kingdom, 19 from Japan, 18 from Russia, 14 from France, and 10 from other participating countries. The papers were presented in both oral and poster sessions over a four-day period. Table 1 provides a listing of the technical sessions for ICNC'95. Each of the "pillar" areas of criticality safety discussed above were represented by a wide range of papers. Discussions on the papers and various general issues were facilitated by the fact that the attendees were together daily to share breakfast and lunch in an informal collegial atmosphere.

Even with the decline in experimental programs in many countries, there were a number of papers that discussed new and planned critical experiments. The first experimental results from the Static Experimental Critical Facility (STACY) in Japan were reported by Y. Miyoshi of the Japan

Table 1. Titles of technical sessions at ICNC'95

Session	Session Title
1A	Current NCS Issues
1B	Burnup Credit, National Programs
2	Data and Validation
3	Codes
4	Overviews of Programs and Facilities
5	Burnup Credit
6	Codes, Data and Benchmarks
7	PRA and Safety
8	NCS Case Studies
9	Standards, Data and Training
10	Solutions, Accidents and Transients
11	Case Studies, Experiments, and Safety
12	Mixed Oxide NCS
13	Emergency Planning and Experiments

Atomic Energy Research Institute. The Russians and French reported on their ongoing critical experiment programs, while the U.S. papers concentrated mostly on needs and plans for future experiments. A paper by B. Briggs of the United States discussed a collaborative project sponsored by the U.S. Department of Energy (DOE) and the Organization for Economic Cooperation and Development (OECD) that is seeking to create an international data base of evaluated, well-characterized critical experiments that have been performed throughout the world over the last fifty years. This effort is expected to continue for several years. Several papers from Japan and the United States centered on subcritical measurements and subcritical techniques in general. The papers indicate that these techniques are being considered and applied to help address a number of criticality safety issues related to production processes, storage, and handling of fissile material. J. Mihalczo and T. Valentine of the United States also had papers that discussed the potential for these techniques to be used to help validate codes and identify cross-section data deficiencies. The papers and discussions at ICNC'95 indicate that the criticality safety community still places a high premium on critical experiment data. This interest is exemplified by the efforts being put forth to retain existing data in easily accessed data bases, the enthusiasm over the new NUCEF facility in Japan, the availability of new data from Russia, and the continued efforts to apply subcritical techniques.

New enhancements to codes and data used for criticality safety analysis were presented. In particular, enhancements and validation experience with the SCALE/KENO, MCNP, and MONK codes were discussed in several papers. Validation of a new multigroup Monte Carlo code from France called TRIMARAN2 was also presented. Unlike previous conferences, there were some extensive presentations and discussions on nuclear data for criticality safety analyses. Some new libraries derived from JEF and ENDF/B-VI were presented. A renewed interest is being generated in an effort to better understand the effect of different nuclear data evaluations and processing methods on k_{eff} of applications that have not been well characterized (e.g., intermediate-energy spectra).

The application of criticality safety data and principles to storage, transport, fabrication processes, disposition, and reprocessing areas were provided in an array of papers throughout the conference. At least two papers from the United Kingdom provided a perspective of the certification authorities towards criticality safety. One important contribution to the conference was the announcement that the Nuclear Criticality Safety Handbook of Japan was available in an English translation. Available copies of the Handbook were picked up quickly by attendees.

A definite trend in operational issues is the increased effort to defend assumptions, operational parameters, and personnel qualifications. This trend is causing increased emphasis to formalize operational guidelines and personnel training. Although the use of probabilistic risk analysis (PRA) methods was mentioned in several papers as a means of helping to defend operational assumptions, the general use of the method does not yet appear to be readily accepted by the community, except on an "as-needed" basis.

One application that provided a focus for the experiment, analysis, and operations areas was burnup credit (i.e., taking credit in the criticality safety evaluation for the reactivity loss in spent fuel). Use of burnup credit requires that those spent fuel isotopics that ensure a bounding k_{eff} be used in the safety analysis. Special operational controls are needed to provide assurance that the fuel has sufficient burnup for the approved application. The lack of critical experiments with spent fuel continues to raise a number of questions relative to code and data validation. All of these areas of burnup credit were reviewed in papers from France, Japan, the United Kingdom, and the United States. Experiments to help validate fission-product cross sections in a spent fuel environment have been performed in the United Kingdom and France. Isotopic validation was the subject of papers from the United States, Japan, France and the United Kingdom. A data base of spent fuel isotopics called SFCOMPO is being developed in Japan. A spent fuel critical experiment planned in the United States was presented. Analysis issues associated with validation and demonstration of accurate, yet bounding, methods were presented by the United States and the United Kingdom. The papers indicate that burnup credit is a new and challenging area for criticality safety specialists, and issue resolution has reached the point that the method can be considered for licensing applications. (The Japanese have had burnup credit accepted for use in their new reprocessing plant that is under construction, and the French are applying actinide-only burnup credit in the transportation of spent fuel.)

CHALLENGES

The ICNC'95 conference provided a useful "snapshot" of the state of the worldwide criticality safety community and the vital issues of interest. A review of the papers indicates several challenges that lie ahead in the coming years. In particular, the criticality safety community will be challenged to:

1. ensure effective and safe implementation of burnup credit;

2. facilitate acquisition and dissemination of the existing Russian data and experience;

3. consider and effectively apply PRA as appropriate;

4. continue reliance on effective linking of experiments, computations, and experience even as pressure mounts to rely more heavily on computations and formality of operations; and

5. improve routes for exchange of measured data, computational software, and experience.

These areas are challenges that the criticality safety community will be addressing between now and the next ICNC meeting that will be held in France in 1999. The authors are confident that the ICNC'99 conference will continue the history of providing an excellent format for the criticality safety community to share experiences and new information that will demonstrate that these challenges are being met.

RELATED MEETINGS

The OECD Nuclear Energy Agency (NEA) is striving to help the criticality safety community meet challenges related to the exchange of information and experience. As part of this effort, the OECD/NEA sponsored a two-day Experts' Meeting on Experimental Needs in Criticality Safety which was held in Albuquerque, New Mexico, the week following the ICNC'95 meeting. The General Chair for the meeting was R. M. Westfall of the United States, and co-chairs were F. Barbry of France and Y. Naito of Japan. The meeting was truly international in perspective, with 28 official participants from 28 organizations in 10 countries. In addition, there were approximately 25 observers.

The stated purpose of the meeting was to review and identify experiments that were deemed necessary to address away-from-reactor criticality safety issues. Representatives from countries that have experimental facilities discussed the status and plans for their facilities and reviewed their areas of expertise as well as the available equipment. Discussions were held on the different needs for experiments that have been identified over the years (large arrays of fissile units, low-moderation systems, etc.) and the applications for which they were pertinent. The recommendations that were agreed to by the participants were:

1. to encourage the performance of new critical measurements on a multilateral, international basis with regard to the sharing of facilities, staff expertise, and funding resources;

2. to establish a Criticality Safety Working Party by the OECD/NEA;

3. to expand the U.S. DOE forecast of criticality experiments to be an international effort;

4. to continue the efforts of the participants and their organizations to obtain base funding for specific critical facilities that could openly provide data needed for the assurance of nuclear safety.

Unfortunately, the goal to define a list of specific experiments that were a priority need was not met. However, the cooperative spirit of the participants and organizations involved in this international meeting was a beneficial step towards cooperative efforts in acquiring new experimental data. It is clear that the OECD/NEA and perhaps the International Atomic Energy Agency (IAEA) must help to organize future meetings, and that these international agencies must play a larger role in ensuring that data for nuclear safety are obtained as needed.

SUMMARY

The ICNC'95 conference was a successful meeting by any measure of performance, that is, applied–participation, number and quality of papers, information exchange, etc. It is obvious that criticality safety issues are of growing interest as the form and quantity of fissile material being handled and stored continues to change and the demand for quantitative justification of safety assessments increases. This conference provided an excellent forum for the exchange of information and ideas that will benefit criticality safety specialists for years to come. Further indication of the criticality safety community working together to address issues of common importance is demonstrated by the Experts' Meeting that was held just following the ICNC conference. Future meetings of this type will continue to help the community to share experiences and knowledge that will benefit the safety of the worldwide nuclear industry.

Proceedings of ICNC'95 and videos of the Pioneer and Russian Plenary Sessions can be obtained at a reasonable cost by contacting R. D. Busch at

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Critical Experiments on Low Enriched Uranyl Nitrate Solution with STACY

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INTRODUCTION

The Static Experiment Critical Facility, STACY in the Nuclear Fuel Cycle Safety Engineering Research Facility, NUCEF. is a solution type critical facility to accumulate fundamental criticality data on uranyl nitrate solution, plutonium nitrate solution and their mixture[1,2].

A series of critical experiments for 10 wt% enriched uranyl nitrate solution using a cylindrical core tank of 60 cm in diameter have been performed with the STACY. The initial criticality was achieved on February 23 in 1995 and various function tests were carried out for obtaining the operation license by Science and Technology Agency (STA) of Japan.

In these experiments, systematic data of the critical height, differential reactivity of the fuel solution, kinetic parameter and reactor power were measured with changing the uranium concentration of the fuel solution from 310 gU/ℓ to 225 gU/ℓ. The differential reactivity of the fuel solution height was obtained by measuring reactor period at a slightly super critical solution height in order to evaluate the nuclear limitations such as maximum reactivity, maximum reactivity addition rate which are fundamental quantity for the safe operation.

Kinetic parameters βeff$/l$, where βeff is effective delayed neutron fraction and l is prompt neutron life time which determine the transient behavior of fuel solution system were also measured by the reactor noise method and the pulsed neutron source method.

The reactor power of the experimental core was calibrated by the gamma ray spectrometry for fission products in the fuel solution which was sampled after high-power operation. Neutron introduction method using a start-up neutron source and the activation method of Au foil were also applied and the availability of these method were confirmed by comparison with the evaluated power by the FP analysis method.

The main parameters which determine the criticality condition of the uranyl nitrate solution are U235 enrichment, uranium concentration, acid molarity and fuel temperature. Systematic experimental data through the first series of experiments for the basic core with a cylindrical tank of 60 cm diameter are reported in this paper for evaluating the accuracy of the criticality safety calculation codes.

Reference calculations of the neutron multiplication factor k_{eff} for the critical condition were made using a neutron transport code TWOTRAN and a continuous energy Monte Carlo code MCNP4A with the Japanese evaluated nuclear data library, JENDL 3.2.

CRITICALITY FACILITY

STACY consists of the core tank containing fuel solution, solution transfer system, fuel storage system, solution adjusting system, and water reflector system[3,4]. The fuel solution is fed from the storage tank to the core tank through a critical approach.

The core tank has a cylindrical geometry with a diameter of 60 cm, which is made of stainless steel SUS 304.

Solution height in the core tank is measured with contact type height gauge. This height gauge has an accuracy of ±0.2 mm. Two B10 counters and four gamma compensated ionization counters were positioned around the core tank to measure in the source range and power range, respectively. The maximum power is limited up to 200 watt. The external neutron source, Am-Be, is inserted below the bottom of the core tank at the beginning of the operation. Reactivity is controlled by adjusting the solution height in the core tank without control rods.

Initially the fuel solution is fed by a fast speed pump system to just below half the predicted critical height. Fuel solution is fed using a slow speed pump system in the near critical state. The maximum excess reactivity and maximum reactivity addition rate are adjusted by limiting the position of the contact type height gauge and the feed speed of the slow pump system. Four cylindrical safety rods containing B_4C pellets are positioned at the upper part of the core tank, and are dropped by gravity in an emergency shutdown condition. The flow diagram of the STACY is shown in Figure 1.

EXPERIMENTS

Main experimental items through the first series of experiments are listed in Table 1.

(1) Critical solution heights

The cylindrical core tank was settled in a reflector pool and the light water was supplied into the reflector pool before the operation. The side reflector and lower reflector are more than 30 cm in thickness. The height of the water reflector was 20 cm more than the upper plate of the core tank. The initial critical approach for a water reflected core was performed on February 23, 1995 using the uranyl nitrate solution containing 310.1 gU/ℓ and 2.1 mol/ℓ free nitric acid with a density of 1.4827 g/cm^3. The uranyl nitrate solution had an enrichment of 9.97wt%. In addition to the two nuclear instruments of B10 counters (Start up channel A and B), four experimental channels composed of three He3 proportional counters and one B10 counter (Channel 1-4) are positioned around the core tank. The critical height was estimated by monitoring the inverse count rate of these neutron counters during subcritical state and the critical height was estimated 41.53 cm from the 11-step measurements. This value was confirmed by observing the steady flux with power range nuclear instruments of two compensated ionization chambers (Linear channel A and B). The fuel temperature was maintained approximately 23 °C.

In addition, the initial critical approach for an unreflected core was made on April 11, 1995 and the measured critical height was 46.8 cm. The uranyl nitrate solution had a concentration of 313 gU/ℓ, 2.3 mol/ℓ free nitric acid with a density of 1.4881 g/cm^3. The detector configuration in the critical approaches is shown in Figure 2 and the reciprocal curves during the subcritical states in the above two experiments are shown in Figure 3 and Figure 4.

In order to obtain the systematic criticality data on the solution height for the 60 cm diameter core tank, the fuel solution was diluted in a storage tank with maintaining the acid molarity about 2.2 mol/ℓ and critical approaches were repeated. The lowest uranium concentrations for the water reflected and unreflected cores were 225 gU/ℓ and 242 gU/ℓ, respectively under the limitation for the maximum critical height of 140 cm. Core conditions in this series of experiments are shown in Table 2, and the measured critical heights are shown in Table 3 and Figure 5.

(2) Tests on nuclear limitations

The nuclear limitations such as maximum reactivity and maximum reactivity addition rate are fundamental quantities for the safe operation of a critical assembly. As the reactivity of the STACY is adjusted by changing the solution height, the differential reactivity as a function of critical height was one of the most important properties. By measuring the reactor period at a slightly super critical state after attaining criticality, this quantity was evaluated using an inhour equation.

Results of the differential reactivities for reflected and unreflected cores are shown in Figure 6. The dependence of differential reactivity on the solution height is determined by the shape of the fundamental mode of the neutron flux along the vertical direction. Fitting curve in this figure is based on the one energy group theory.

In the case of a simple geometry such as cylindrical core, geometrical buckling can be evaluated by fitting flux distribution in the vertical and horizontal direction to each fundamental mode function. The neutron flux distribution along the vertical direction in the reflector region was measured by scanning a He-3 counter using a driving mechanism. The Activation distributions of gold wires attached on the outer surface of the core tank were also measured with a NaI(Tl) scintillation counter using a beta-gamma coincidence counting technique.

In order to evaluate the shutdown margins in both conditions of all-rod insertion and one-rod stuck insertion, the reactivity worth of safety rods were measured using a rod drop method. The decrease in the counting rate of the start up channels A and B after rod insertion were monitored using a multichannel scaler with 4096 channels. The dwelling time of each channel was 0.1 seconds. Examples of the measured and calculated neutron multiplication factors after one-rod stuck insertion of the safety rods are shown in Figure 7. Calculations were made by the Monte Calro Code MULTI-KENO with the nuclear data library JENDL 3.2[5].

(3) Power calibration

Reactor power of the core was increased step by step, and the response of the neutron detectors of the start-up channel, linear channel and safety channel were measured. Power calibration was made using an analysis of fission products in the fuel solution which was sampled after the operation. The gamma activities of fission products such as Ce143 and Ba140, and Np239, which is produced by capture reaction of U238, were measured with a Germanium detector.

The activity was corrected to that at the shutdown time by taking account of the half life of each nuclei. An example of the decay for Ce143 and Np239 is shown in Figure 8. Results of the reactor power based on the fission products and Np239 is shown in Table 4.

The neutron introduction method using the start-up neutron source was also tested and the correlation between the gradient of the power increase and the initial power level was obtained. An example of the measurement using the multichannel scaler for start up channel A is shown in Figure 9. The dwelling time of MCS is 1.0 sec and total channel number was 4096.

In addition, the activation method of Au foil attached to the outer surface of the core tank was applied to the experimental core, and the validity of these methods were confirmed by comparison with the calibrated power by FP analysis method. In the irradiation of gold foil, the integral power of the operation was estimated by combining the activity of the Au foil and activation rate calculated by MCNP4A, a continuous neutron Monte Carlo code[6].

(4) Kinetic parameters

Two different experiments were performed to measure the ratio of effective delayed neutron fraction to the prompt life time. In a noise experiment of break frequency method, the current signal from two compensated ionization chamber (Channel 5 and Channel 6) were converted to voltage signal and stored using analog data recorder. Auto power spectrum density, cross power spectrum density and coherence function between the two channels were measured using the fast fourier analyzer at critical state.

In the pulsed neutron experiments, a pulsed neutron source was positioned in the reflector region near the core tank. The change in the count rate of B10 counters after injection of pulsed neutrons were measured in some sub-critical states. The subcriticality was adjusted by changing the solution height within approximately 5 dollars. The pulse rate was 20 per second and total number of induced pulses was approximately 10000. The dwelling time of the multichannel scaler was 100 micro seconds. The results of the kinetic parameters $\beta eff/l$ for the cores with a uranium concentration of approximately 310 g/ℓ are shown in Table 5.

CALCULATIONS

(1) Calculation method

Before the initial critical approach, the critical solution height was estimated using the SRAC code system developed at JAERI[7]. The critical concentration of uranium as a function of solution height was calculated using two-dimensional transport code TWOTRAN. The number of neutron energy group was 16, which was condensed using a one dimensional transport code ANISN. The fine group cross section data of the SRAC is based on the JENDL 3.2 nuclear data library. The value of keff used for searching the critical height was 1.006 with assuming the acid molarity of 2.0 mol/ ℓ.

The main parameters which determine the criticality condition of the solution system of uranyl nitrate are U235 enrichment, uranium concentration, acid molarity and fuel temperature. The sensitivities of the reactivity to these parameters and core dimensions are important to estimate the accuracy of the experimental results. A sensitivity analysis of the neutron multiplication factor for the above parameters was performed for typical experimental cores. The uncertainties of experimental keffs in the critical state for both the water reflected core and unreflected core are summarized in Table 6.

(2) Reference calculations

Some calculations of neutron multiplication factor keff for the critical conditions were conducted using a deterministic transport code, TWOTRAN and a continuous energy Monte Carlo code MCNP4A. The calculation models for each code is shown in Figure 10.

When using the Monte Carlo code, a three dimensional model including the water reflector above the core tank was adopted. On the other hand, a simple RZ model was employed in TWOTRAN, which did not include the upper part of solution height. Distribution of the calculated Keff for reflected cores and unreflected cores are shown in Figure 11 and Figure 12, respectively.

In these calculations, the atomic number densities of the main nuclides in the fuel solution were estimated based on the measured density of the fuel solution. The criticality calculations based on the atomic number densities which were derived from the SST density formula were also performed with the TWOTRAN code.

The maximum difference between the calculated keff and the experimental results (1.0) are within 1.0 % taking account of the experimental error and the calculation results show that the SST density formula is suitable for determining the neutron multiplication factor in the criticality safety analysis[8,9].

SUMMARY

As the STACY started steady operations, systematic criticality data on low enriched uranyl nitrate solution system could be accumulated. Main experimental parameters for the cylindrical tank of 60 cm in diameter were uranium concentration and the reflector condition. Basic data on a simple geometry will be helpful for the validation of the standard criticality safety codes, and for evaluating the safety margin included in the criticality designs.

Experiments on the reactivity effects of structural materials such as borated concrete and polyethylene are on schedule next year as the second series of experiments using 10 wt% enriched uranyl solution. Furthermore, neutron interacting experiments with two slab tanks will be performed to investigate the fundamental properties of neutron interaction effects between core tanks. These data will be useful for making more reasonable calculation models and for evaluating the safety margin in the criticality designs for the multiple unit system[10,11].

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Table 1 Main items of reactor physics test and their requirements

No.	Experimental Item	Limit value	Measurement	Instrumentation
1	Critical height	40-140 cm	Inverse multiplication	B10 counter, He3 counter
2	Maximum reactivity	0.8$	Reactor period measurement	Compensated Ionization Chamber (CIC)
3	Maximum reactivity addition rate	3 cent/sec		
4	Reactivity of driving mechanism	≦30 cent		
5	Shutdown margin at all-rod insertion	Keff≦0.985	Rod drop method	B10 counter, He3 counter
6	Shutdown margin at one-rod stuck	Keff≦0.995		Multi-Channel Scaler (MCS)
7	Reactor power	Power ≦200 watt	Neutron source introduction method	B10 counter ,CIC, Am-Be source
			Gamma activity of fission products and Np239	Ge(Li) detector, γ -ray spectroscopy
			Activation of gold foil	NaI(Tl) Scintillation , β - γ coincidence system
8	Kinetic parameter β eff/l	-	Reactor noise method	Fast Fourier Analyzer, CIC, Analog Data Recorder
		-	Pulsed neutron method	Pulsatoron, B10 counter, MCS
9	Neutron flux	-	Activation of gold wire	NaI(Tl) Scintillation counter
		-	Scanning of neutron counter	Counter driving system

Table 2 Experimental condition for a cylindrical tank of 60 cm in diameter at STACY

Fuel solution : Uranyl nitrate solution	
U235 enrichment (%)	9.97
Uranium concentration (g/l)	225.3 - 317.4
Acid molarity (mol/l)	2.17 - 2.28
Temperature (℃)	23.1 - 25.9
Core tank : SUS304	
Type	Cylinder
Inner diameter (mm)	590
Thickness of side wall (mm)	3
Thickness of bottom plate (mm)	20
Thickness of upper plate (mm)	25
Reflector : Light water* / None (bare)	
Thickness of lower reflector* (mm)	330
Thickness of upper reflector* (mm)	200
Thickness of side reflector* (mm)	> 700

Table 3 Measured critical heights for the cylindrical core of 60 cm in diameter

Run No.	Reflector	Uranium concentration (g/l)	Acidity (moll/l)	Core Temperature (℃)	Critical Height (cm)	Density (g/cc)	Date	Sampling date
1	Water	310.1	2.17	23.1	41.53	1.48266	1995/2/23	1995/2/22
12		312.3	2.22	23.3	41.22	1.48640	1995/4/4	1995/4/3
28		317.4	2.25	25.2	40.94	1.49289	1995/5/25	1995/5/26
29		290.4	2.23	24.8	46.70	1.45717	1995/5/30	1995/5/26
33		270.0	2.20	24.7	52.93	1.43479	1995/6/9	1995/6/12
34		253.6	2.24	24.8	64.85	1.40902	1995/6/12	1995/6/12
46		241.9	2.27	24.6	78.56	1.39357	1995/7/6	1995/7/4
51		233.2	2.28	22.4	95.50	1.38480	1995/9/20	1995/9/19
54		225.3	2.28	23.3	130.33	1.37220	1995/9/26	1995/9/25
14	None	313.0	2.25	23.8	46.83	1.48807	1995/4/11	1995/4/10
30		290.7	2.23	25.4	54.20	1.45711	1995/6/1	1995/6/2
32		273.1	2.24	25.8	63.55	1.43389	1995/6/7	1995/6/2
36		253.9	2.23	25.8	83.55	1.41018	1995/6/21	1995/6/20
49		241.9	2.27	23.4	112.27	1.39410	1995/7/13	1995/7/12

* Measured at 25 ℃

Table 4 Reactor power calibration based on gamma ray activity of fission products and Np239 after a reactor opeartion

Date	1995/3/9	1995/4/25
Run No.	6	19
Reflector	Water	None (bare)
U concentration (g/l)	311.1	314.1
Acid molarity (mol/l)	2.21	2.23
Critical height (cm)	41.48	46.84
Fission density (fission/cc)		
Ce143	1.008×10^9	3.104×10^9
Ru103	1.012×10^9	3.221×10^9
Ba140	9.738×10^8	3.286×10^8
Zr 95	9.484×10^8	3.951×10^9
Np239	9.125×10^8	2.938×10^9
Fission (U235+U238) / Capture (U238)*	6.204 ± 0.002	6.070 ± 0.002
Reactor power based on Ce143 (W.min)	210.2	647.2

* Calculated value with Monte Carlo code, MCNP 4A with the Japanese Evaluated Nuclear Data Library JENDL 3.2

Table 5 Measurement of kinetic parameters β eff / l

Core condition	Date	1995/5/25	1995/4/20
	Run No.	28	18
	Reflector	Water	None (bare)
	U concentration (g/l)	316	313.8
	Acid molarity (mol/l)	2.20	2.23
	Critical height (cm)	40.94	46.83
	Temperature (℃)	25.2	23.8
β eff /l	Pulsed neutron method Ch. A Ch. B	 136.2±4.5 137.9±5.7	 133.6±4.9 135.1±3.2
	Reactor noise method CPSD (Ch. 4,5)	139.4±3.6	134.1±3.2

* Cross Power Spectral Density between CH-4 and Ch-5 (CIC)

Table 6 Uncertainties of experimental keffs for the initial water reflected core and unreflected core

Parameter [p]	Uncertainty of measurement	Reflected core (Run No. 1)		Uneflected core (Run No. 14)	
		Sensitivity (%Δkeff/Δp)	Uncertainty of keff (%Δkeff)	Sensitivity (%Δkeff/Δp)	Uncertainty of keff (%Δkeff)
235U enrichment (%)	± 0.013 a)	0.35 / 0.1 %	0.046	0.35 / 0.1 %	0.046
Fuel concentration (gU/l)	± 0.5 b)	0.06 / 1.0 g/l	0.030	0.05 / 1.0 g/l	0.025
Acid molarity (mol/l)	± 0.02 c)	-0.21 / 0.1 mol/l	-0.042	-0.22 / 0.1 mol/l	-0.044
Temperature (℃)	± 0.3	-0.04 / 1.0 ℃	-0.012	-0.04 / 1.0 ℃	-0.012
Inner diameter (mm)	± 1.5	0.04 / 1.0 mm	0.060	0.05 / 1.0 mm	0.075
Solution height (mm)	± 0.2	0.05 / 1.0 mm	0.009	0.04 / 1.0 mm	0.007
Impurity (Fe,Cr,Ni)	(< 40)	0.005 / 60 ppm	0.003	0.01 / 100 ppm	0.002
Total			0.093		0.102

a), b), c)· Relative error of the measurement
-235U enrichment <0.13 %, U concentration < 0.2 %, Acid molarity < 1.0 %

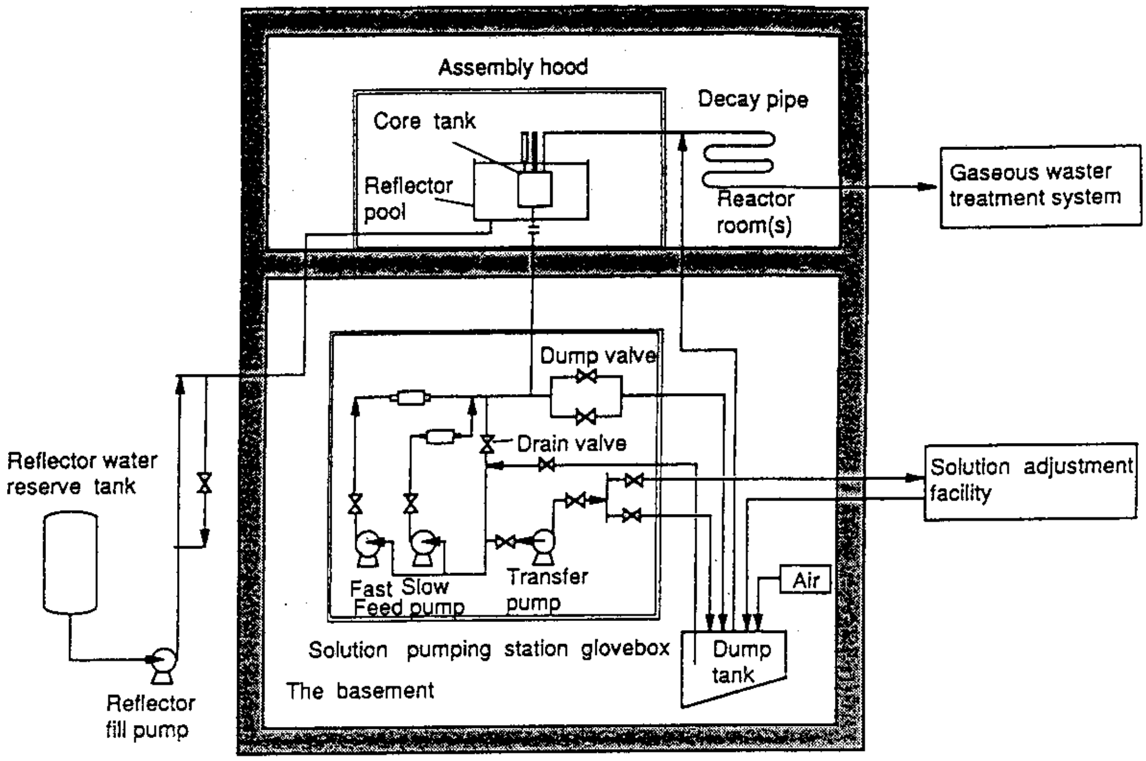

Figure 1. Outline of STACY facilities and flow diagram

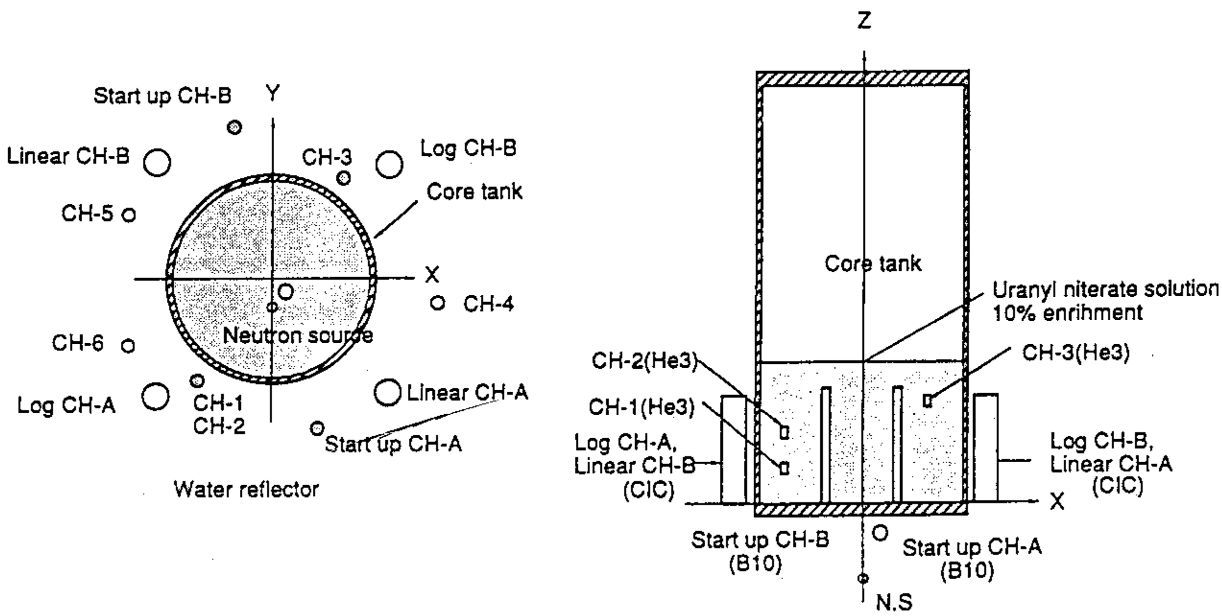

Figure 2 Configuration of neutron counters positioned around the core tank

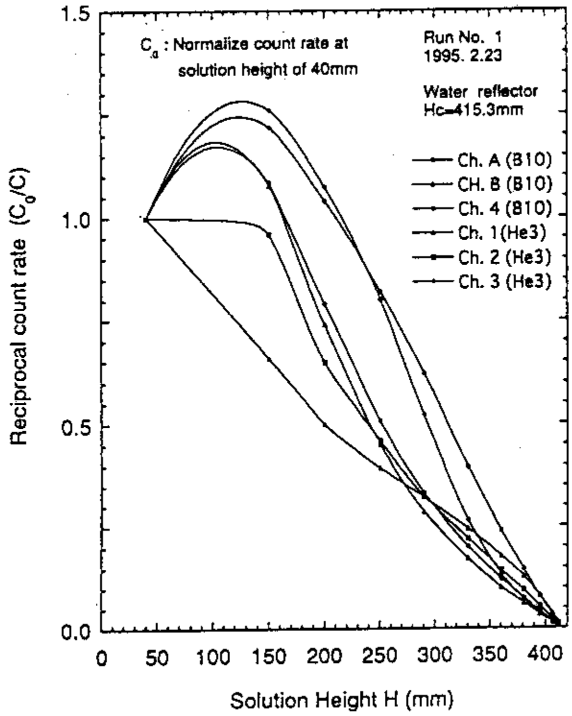

Figure 3. *Reciprocal multiplication in critical approach for a water reflected core*

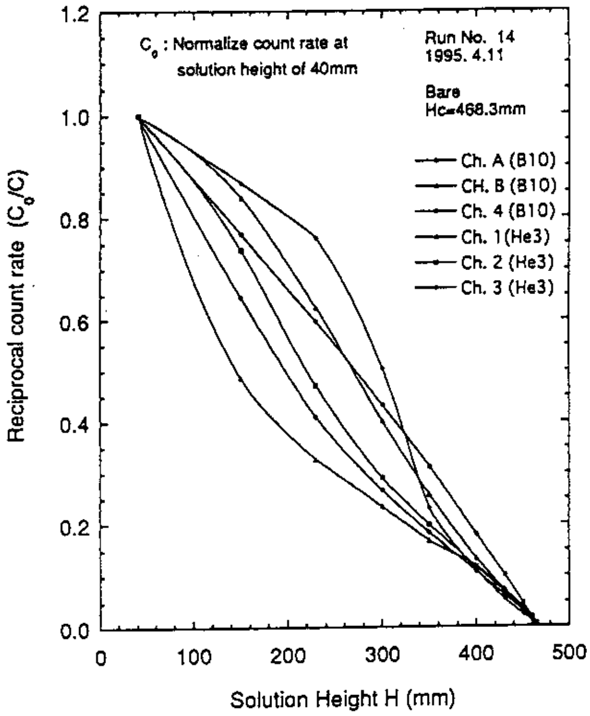

Figure 4. *Reciprocal multiplication in critical approach for an unreflected core*

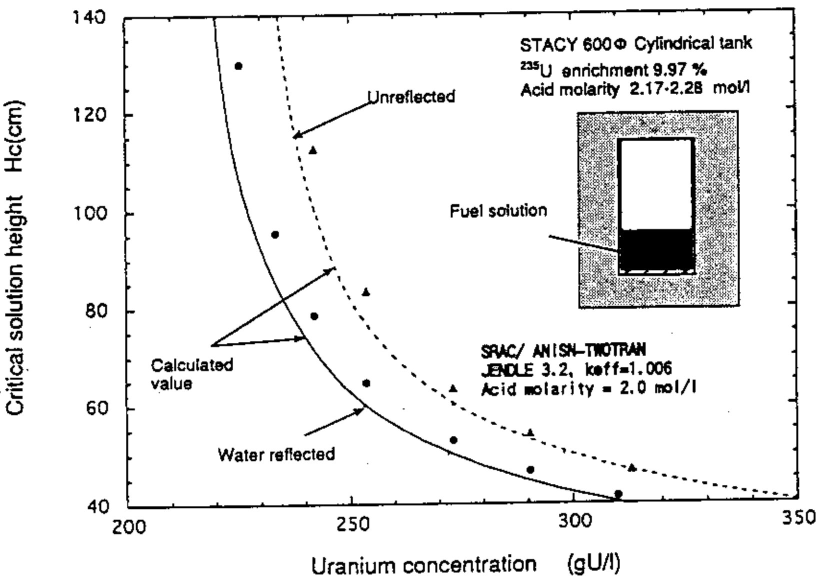

Figure 5. Measured critical solution heights of 10% enriched uranyl nitrate solution in a cylindrical tank

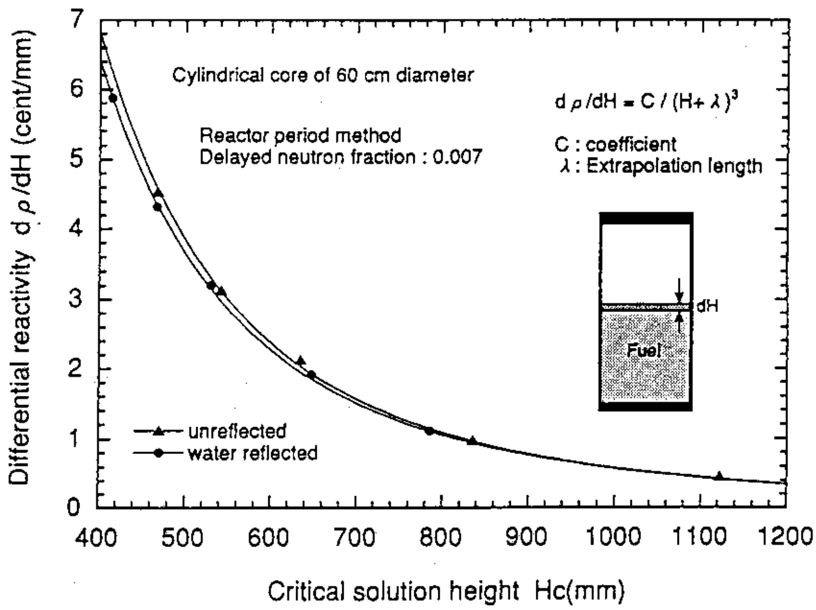

Figure 6. Differential reactivity of solution height

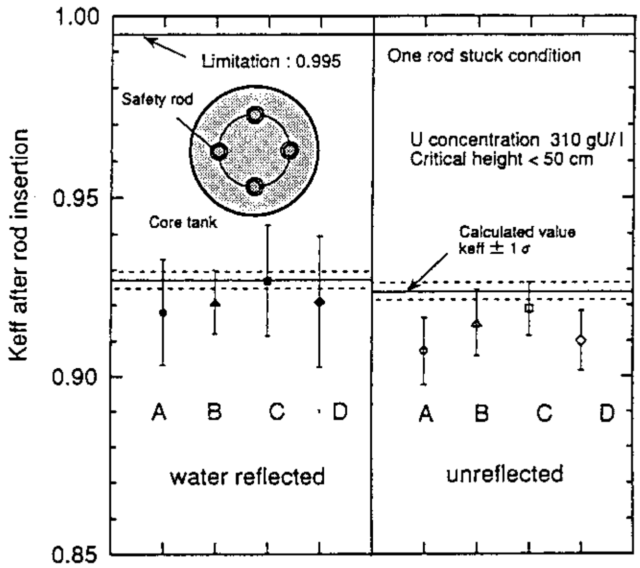

Figure 7. Neutron multiplication factor after one rod stuck insertion of the safety rods

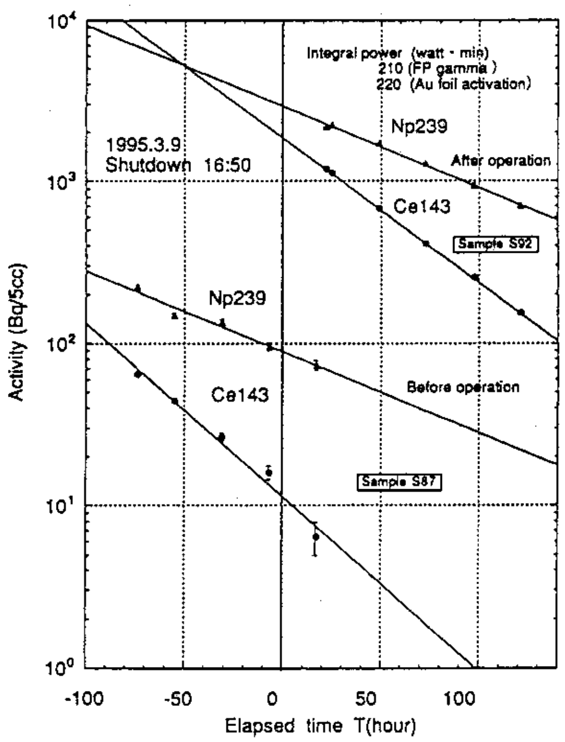

Figure 8. Decay of the activities of Ce143 and Np239 in the fuel solution

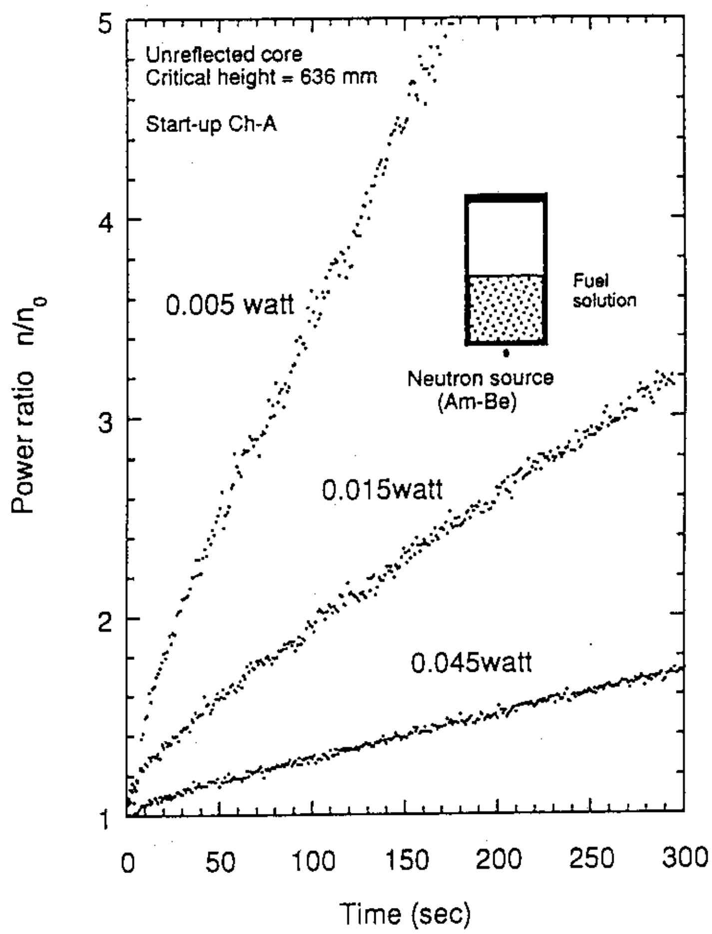

Figure 9. Neutron source introduction method

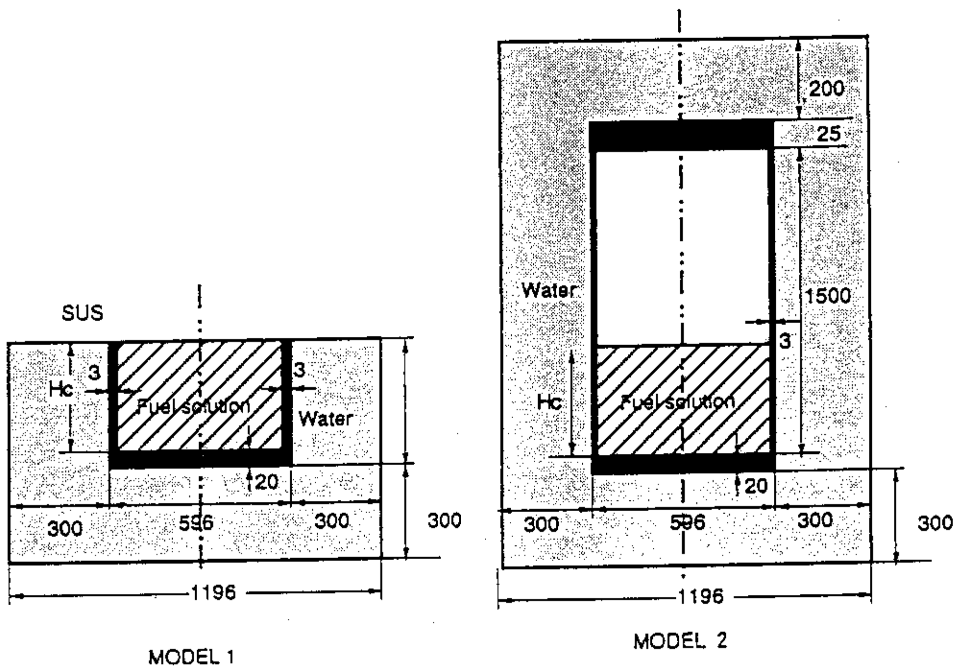

Figure 10 Calculation models

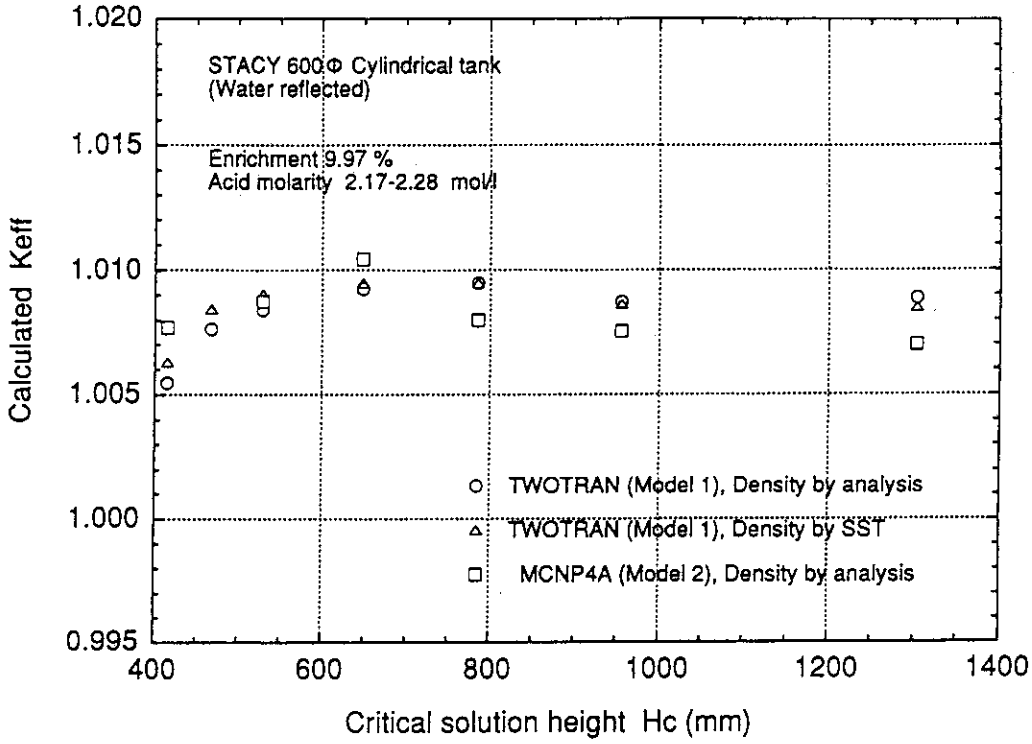

Figure 11. Calculated keff of a cylindrical core of 60cm in diameter with water reflector

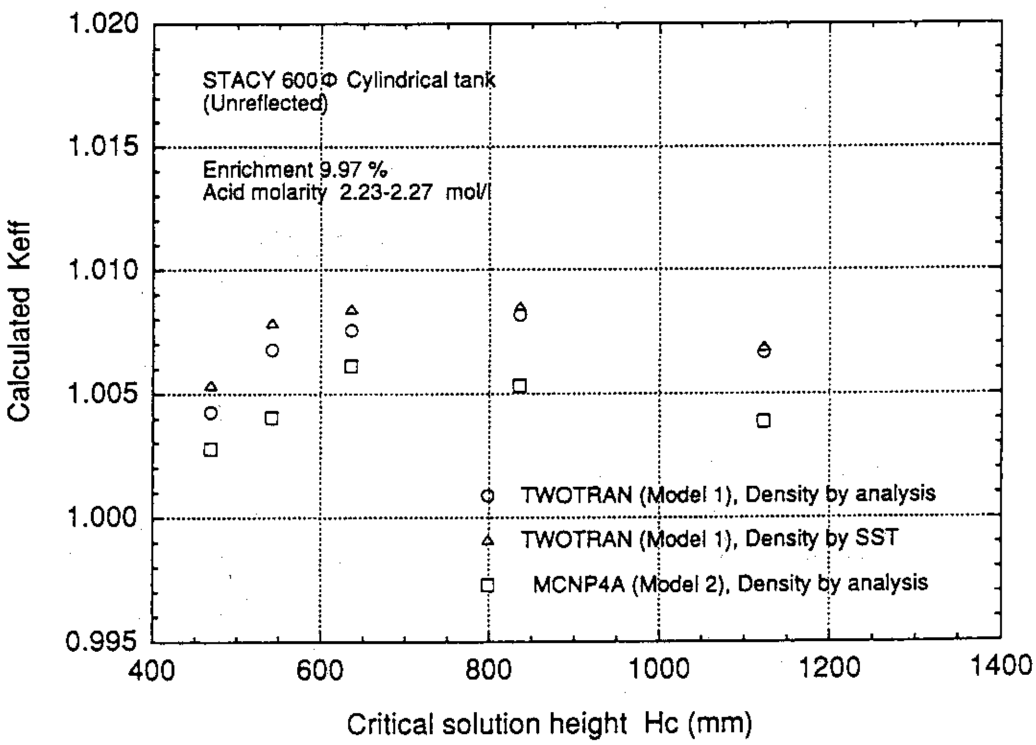

Figure 12. Calculated keff of a cylindrical core of 60cm in diameter without reflector

Tuesday, October 17th, 1995

Technical Session (2)

Accident Evaluation and Probabilistic Safety Analysis

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A Safety Evaluation of Fire and Explosion in Nuclear Fuel Reprocessing Plants

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Abstract

The demonstration test was performed in JAERI to prove the adequacy of a safety evaluation for an air−ventilation system in the case of solvent fire and red−oil explosion in a nuclear fuel reprocessing plant. The test objectives were to obtain data of the safety evaluation on a thermofluid behavior and a confinement effect of radioactive materials during fire and explosion while the system is operating in a cell. The computer code was developed to evaluate the safety of associated network in the ventilation system and to estimate the confinement of radioactive materials in the system. The code was verified by comparison of code calculations with results of the demonstration test.

1. Introduction

Since TBP (tributyl phosphate) with nitric acid is used in the reprocessing plant, the safety evaluation has something in common with that of chemical plants. However, the reprocessing plant is designed to confine radioactive materials effectively by the ventilation system consisting of cells ducts, dampers, HEPA (high efficiency particulate air) filters and blowers, when radioactive materials would be transferred to the system during the accidents. One of the most important confinement system under such conditions consists of HEPA filters installed at the end of the ventilation system.

To prove the adequacy of safety evaluation picked out as hypothetical fire and explosion accidents, the demonstration test of following items has been performed: (1) solvent fire tests to examine fire behabior and the confinement of radioactive materials in the ventilation system, (2) solid−state rocket fuel burning tests to obtain transient gas dynamic behavior propagating through the network, and (3) explosion tests of red−oil formed by a nitration for TBP with nitric acid. Figure 1 shows a schematic of large−scale test facility, which consists of a 20m ^{3} −first model cell and a 6m ^{3} −second model cell: they are 1/4 size of cells in a reference fuel reprocessing plant, 10m− and 50m−long second ducts with 0.2m ϕ in dia., a real−size filter bank to maintain the normal flow rate of 4380 m ^{3} /h with two stages of the six half−size HEPA filters. Small−scale tests were also carried out to elucidate a release of radioactive materials during the fire and a thermal decomposition behavior of solvent with nitric acid under an exothermic reaction of the nitration and chemical form analysis of red−oil in the nitrated solvent.

2. Solvent Fire Test

The solvent fire test was carried out by the test facility focusing on the confinement of radioactive materials in the ventilation system [(1)−(4)]. The solvent used in the tests was 30vol.%TBP/n−dodecane on water phase containing fission product simulants of Cs, Sr, Ce, Ru and U. Table I shows major fire conditions and results. In the tests, the fire behavior was observed under different ventilation flow rates and different burning areas of solvent, because these two parameters were considered to have significant effects on fire behavior. The burning area of solvent ranged from 0.0768 to 0.9216m ^{2}, which corresponds to a 1/4 pool surface area of 1 to 12 stages of mixer-settlers. Measurements were made of temperatures, pressures, flow rates in cells and ducts, pressure drop across the HEPA filter, and contents of burning gases in the first model cell. Samples of smoke containing fission product simulants were taken by Maypack-type samplers at the first and second ducts. The amount of fission product simulants trapped in samplers was determined by an activation analysis. Figure 2 shows burning rates of solvent over times corresponding to burning areas of 1 to 9 stages of mixer-settlers under the flow rate of 30 air-volume changes per hours. In the figure, burning rates were proportional to the number of stages during a steady-state burning and showed almost constant, but the burning rates increased rapidly at the end of the fire due to an explosive burning called as the boilover burning of solvent vapor in the cell atmosphere at a sudden boiling of water under solvent phase, and fire quenched by a lack of oxygen in the cell.

Following results were obtained in the fire demonstration test. (1) A combustion region of 30vol.%TBP/n-dodecane proved to be narrow compared with n-dodecane burning in the tests. Even under the combustion region, an extinguishment of the fire is made when the boilover burning begins in the cell. (2) The burning rates of solvent were determined under the wide range of ventilation flows and burning areas. These burning rates are in quite agreement with calculations by a solvent pool fire model taking into consideration a heat feedback from flame to the pool surface and a diffusion of oxygen from atmosphere to the flame [5], [6]. (3) The smoke generation rate was 5 ~ 10% of the burning rate. A particle size distribution of smoke was measured as D_g =0.13 to 0.18 and σ_g =1.6 to 2.1. (4) The deposition behavior of smoke along ducts was governed by thermophoresis of aerosol from hot gas to the cold wall. (5) The release rate of radioactive materials from burning solvent was determined in the tests. It is important to extinguish forcibly the fire before water boiling because the release of radioactive materials becomes large during the boilover burning. (6) Smoke plugging of the HEPA filter and DF-value were measured in the tests. Figure 3 shows the pressure drop across the HEPA filter against the cumulative weight of burned solvent per one unit of the half-size HEPA filter. In the test, the filter would suffer almost no effects from heat transport and smoke plugging, and the overall DF can be maintained at ~ 10^5 for smoke. In the demonstration test, the FACE code was developed to evaluate a safety ability of the ventilation network during the fire and verified by comparison of code calculations with results of the test [7].

3. Explosion Test

It is important for the safety evaluation that integrity of the HEPA filter is maintained during the explosion accident. To determine mitigating effects of temperature, pressure, and flow rate in the network during the explosion, the rocket-fuel burning test was carried out by the burning of a solid-state rocket fuel made of mixed nitrocellulose and nitroglycerin [8]-[10]. Table II shows major test conditions. Figure 4 shows measured pressures along cells and ducts in the test facility under the burning of 6kg-rocket fuel for 5s in the first model cell. In this test, heat and gas release rates into the cell atmosphere were 4.4 MJ/s and 1.04 m ^{3} /s. Figure 5 shows the decline of propagated pressure peaks for the rocket-fuel burning tests with distance from the bottom of the first model cell to the outlet of the HEPA filters. It is notable that there are significant pressure decreases due to a fluid resistance at bent ducts, sudden expanded and sudden contracted joints among cells and ducts, confluences of ducts, and dampers. From these results, mitigating effects were ascertained due to structures of the network.

To understand sensitive decomposition materials in the nitrated solvent, chemical analysis of red-oil was performed in the tests. Figure 6 shows nitrated-compounds measured with the nitrogen atom detector of a gas chromatograph for 30vol.%TBP/n-dodecane with nitric acid heated at 145℃. From the analysis, sensitive decomposition materials too weak heating were proved to be butyl nitrite and butyl nitrate formed by the reaction of TBP with nitric acid. In the tests, reaction kinetic rate and reaction heat of nitric solvent during the thermal decomposition were also measured by the small-scale tests [11].

To confirm an extent of the red−oil explosion, the explosion test was carried out in the first model cell using by a sealed vessel containing TBP, n−dodecane and nitric acid at a constant heating rate under operating ignition plugs in the cell. Table III shows major test conditions. Figure 7 shows solvent temperatures in the vessel under different volume ratios of TBP to nitric acid and different concentrations of nitric acid. In the tests, a rapid temperature increase occured about 145 ℃ by the exothermic reaction between TBP and nitric acid, and a rapture disk on the vessel burst at the fixed pressure by an accumulation of thermal decomposition gases. Simultaneously, explosion occured in the cell by the ignition of solvent mist and flammable decomposition gases. Figure 8 shows pressures in the first model cell for red−oil explosions and the rocket fuel burning. By comparing the integrated value of pressure curves between these explosion tests and the rocket fuel burning test, maximum heat release rate of the explosion is 970kJ/s at a top of the curve for the explosion of 1kg−nitrated solvent and its maximum gas release rate is $0.22m^3/s$. The integration of these heat and gas release rates is found to be nearly equal to that of the rocket fuel burning. Figure 9 shows the decline of pressure peaks of red−oil explosions and the rocket fuel burning at various points between the first model cell and the HEPA filter.

4. Basic Equations of Computer Code

For analyzing the safety evaluation, it is necessary to calculate accurately thermofluid behavior in the ventilation network taking into consideration basic equations of continuity, momentum, energy, and state; mass transport behavior of aerosol and burning gases containing radioactive materials; pressure loss for cells and ducts in the network; heat and mass release rates in the cell as source terms; heat transfer to the wall of cells and ducts; deposition behavior of aerosol onto the wall; filtration of the HEPA filter and aerosol plugging during fire and explosion. Table IV shows basic equations in a CELVA−1D code, which is an one−dimensional compressible code to calculate thermofluid dynamics and mass transport in the network by a node−junction method. In the node−junction method, pressure, temperature, density and mass concentration are calculated in the node, and velocity is calculated at the junction. Table V shows analytical functions and mathematical models of CELVA−1D. To verify the mathematical models, code calculations were compared with results obtained by the demonstration test.

5. Modeling of the Code

In the analysis, thermofluid calculations were obtained by Eqs.(1)−(4) in table IV by difference calculus with the Newton's method, and mass transport calculations were by Eq.(5) involving a filtration of the HEPA filter. The mass release rate of burning gases was given by $\dot{M}$ in Eq.(1) and the heat release rate by $\dot{Q}$ in Eq.(3). The source term of aerosol containing radioactive materials was given by $\dot{S}$ in Eq.(5) as the mass generation rate. These source terms were already obtained by the demonstration test. The plugging effect by smoke on the HEPA filter was also calculated by Eq.(6).

The drag force in Eq.(2) for the fluid resistance of ducts in the network was given by $F=k\rho v^2/2$; where k is the pressure loss coefficient obtained by k=fL/D; f is the friction factor, L/D the ratio of length to diameter of straight ducts. In straight ducts, the friction coefficient was given by f=64/Re for the laminar and $f=0.1[1.46(\varepsilon/D)+(100/Re)]^{0.25}$ for the turbulent; where Re is the Reynolds number and ε the wall roughness on the inner wall of ducts. For bent ducts, the pressure loss coefficient was given by $k=0.0315(R_0/D)+0.21(R_0/D)^{-0.5}$ for $R_0/D > 1.0$; where R_0 is the curvature of ducts. The pressure loss coefficient at sudden expanded and sudden contracted joints was given by $k=\phi[1-(A_1/A_2)^2]$; where ϕ is a constant value as $\phi=1.0$ for the enlargement and $\phi=0.4$ for the reduction, A_1 and A_2 are cross sectional areas at the inlet and outlet of the joint. Consequently, an overall drag force between upper cell and lower cell in the network can be obtained by the summation of $F_1+F_d+F_2$; where F_1 is the drag force between upper cell and the duct, F_d at the duct, and F_2 between the duct and lower cell. For dampers, the drag force was determined from the relation between pressure drop at the damper and flow rate changing by an open area of the damper. Drag force of the HEPA filter was given by $F=\triangle P_0 (1+\alpha L_0+\beta L_0^2)$ taking into consideration the smoke plugging; where $\triangle P_0$ (Pa) is the pressure drop across a new HEPA filter without plugging. This pressure drop was given by $\triangle P_0 (\mathrm{Pa})=K_L \mu v+K_T \rho v^2/2$; $K_L=5.8\mathrm{x}10^8 (\mathrm{m}^{-1})$ and $K_T=7.96\mathrm{x}10^4 (-)$ were obtained by the demonstration test. The mass weight of smoke loaded on the HEPA filter is given by L_0 (kg) in Eq.(6); plugging coefficients were determined by $\alpha=138 (\mathrm{kg}^{-1})$ and $\beta=3690 (\mathrm{kg}^{-2})$ from L_s (kg/unit) in Fig.3. Drag force of the blower was obtained by the measurement for pressure drop against flow rate through the blower.

Because junctions have not wall surface in the node−junction method, heat transfer to the wall in Eq.(3) is given by $\dot{q}_{w,i}=h_i (T_{g,i}-T_{w,i})+\varepsilon_g \sigma (T_{g,i}^4-T_{w,i}^4)+\varepsilon_f \sigma (T_f^4-T_w^4)$ in the node; where h_i is the heat transfer coefficient, σ the Stefan−Boltzmann coefficient, $T_{g,i}$ and $T_{w,i}$ are temperatures in fluid and on the wall in the ith node, and T_f and T_w are temperatures in flame and on the wall surface in the accident cell. ε_g is the radiative heat emissivity from hot gas to the wall and ε_f from flame to the wall. These emissivities were given by the band model of heat radiation using data of CO_2 and H_2O contents in cells and ducts. The temperature on the wall surface was calculated by a partial differential equation of heat conduction into the walls.

It is assumed that aerosol is transferred to the network with fluid, and is deposited on the wall and finally trapped on the HEPA filter. In Eq.(5), deposition rate of the kth aerosol was given by $\dot{R}_{i,k}=A_i v_{p,i,k} C_{i,k}$ in the ith node, where $v_{p,i,k}$ is the deposition velocity of the kth aerosol toward the wall, A_i the wall surface area and $C_{i,k}$ the concentration of the kth aerosol in the ith node. The deposition velocity was expressed by $v_{p,i,k}=v_{s,i,k}+v_{d,i,k}+v_{T,i,k}$ due to settling, diffusion and thermophoresis of the kth aerosol. In Eq.(5), the deposition efficiency of aerosol at each junction is $\eta=0$ without the HEPA filter because the junction possesses no wall surface area expect the filter in the node−junction method. The weight of aerosol loaded on the HEPA filter (L_0) was obtained from Eq.(6) by the summation of C_k before the filter and flow velocity (v) through the filter. At the HEPA filter, the value of η_k for the kth aerosol was calculated by the Kirsh's model of filtration efficiency [12].

6. Code Verification by the Tests

To ensure mathematical models in CELVA−1D, verifications of the computer code were carried out by data of the demonstration test. Figure 10 shows temperature curves during the solvent fire calculated by CELVA−1D at the first model cell and the second duct, and these measurements by the fire test under the burning of 1 stage mixer settler with the flow rate of 30 air−volume changes per hours. In the figure, relatively good agreements are obtained against measured curves, even though boilover burning occurs in the cell. Figure 11 shows the measured propagation of temperatures along the network of the test facility in the rocket−fuel burning test under the burning of 6kg−rocket fuel for 5s with the flow rate of 6 volume changes per hours. In this test, an inserted pipe (0.09m ϕ x1m) was attached in the 0.2m ϕ −second duct as the fluid resistance. Figure 12 shows the decline of pressure and temperature peaks calculated by CELVA−1D with measured values in Fig.11 against distance from the first model cell to the HEPA filter. These calculated declines are in good agreement with measured values. In the figure, a significant pressure decrease occurs at the inserted pipe due to the fluid resistance, and a significant temperature decrease appears at the second model cell due to an expansion of fluid.

To elucidate the confinement of radioactive materials during a fire in the reference reprocessing plant, code calculations for the safety evaluation were performed by CELVA−1D. As results, integrity of the HEPA filter at the end of the ventilation system was maintained for heat transport and smoke plugging, even if boilover burning occurred in the cell. The red−oil explosion at an evaporator in the fuel concentration process of the reprocessing plant was also calculated by CELVA−1D to elucidate the safety of the evaporator and mitigating effects by an off−gas ventilation network in the process. From the calculation, it is found that the evaporator has an enough safety margin for the explosion and the safety of the ventilation system is kept enough by the effective mitigation of the network. Also, integrity of the HEPA filter would suffer almost no effect for the explosion.

7. Conclusion

Many test data for the safety evaluation in the case of fire and explosion were gained by the demonstration test. It is important to estimate accurately the thermofluid behavior in the network, because the confinement of radioactive materials is closely governed by the thermofluid. CELVA−1D code was developed and verified by the demonstration test focusing on thermofluid behavior and mass transport phenomena. The code can be calculated not only the thermofluid behavior in the network, but also the mass transport of radioactive materials including integrity of the HEPA filter. So that, CELVA−1D will be able to use calculations of the safety evaluation for a fuel reprocessing plant within a short computation time, even though the plant has a complicated ventilation network.

Nomenclature

A : Cross sectional area of nodes at cells and ducts in the network (m^2)
A_w : Inner surface area of nodes at cells and ducts in the network (m^2)
C : Concentration of aerosol (kg/m^3)
C_v : Specific heat (J/kgK)
D : Diameter of straight duct (m^2)
DF_k : Decontamination factor of the kth aerosol by HEPA filter (−)
e : Internal energy of fluid gas in node (J/kg)
F : Drag force of junctions by flow resistance of cells ducts, dampers, filters and HEPA filter in the network (Pa)
f : Friction factor (−)
h : Heat transfer coefficient ($J/m^2 s ℃$)
K_L : Filter coefficient for laminar flow (m^{-1})
K_T : Filter coefficient for turbulent flow (−)
k : Pressure loss coefficient (−)
L : Length of straight duct (m)
L_a : Weight of aerosol loaded on the HEPA filter (kg)
L_s : Weight of aerosol loaded on one unit of the half−size HEPA filter (kg/unit)
$\dot{M}$: Mass release rate of burning gases as source term during fire and explosion (kg/s)
P : Pressure in nodes in the network (Pa)
ΔP_0 : Pressure drop across a new HEPA filter (Pa)
$\dot{Q}$: Energy release rate of burning gases as source term during fire and explosion (J/s)
q : Artificial pressure for stability (Pa)
$\dot{q}_w$: Heat flux from fluid to the wall of cells and ducts ($J/m^2 s$)
$\dot{R}$: Deposition rate of aerosol at nodes ($kg/m^2 s$)
R_0 : Curvature of duct (m)
Re : Reynolds number ($=Dv\rho/\mu$)
$\dot{S}$: Generation rate of aerosol at node as source term ($kg/m^2 s$)
T : Temperature (℃)
t : time (s)
v : Flow velocity of fluid at junction (m/s)
v_d : Deposition velocity of aerosol due to diffusion (m/s)
v_s : Deposition velocity of aerosol due to settling (m/s)
v_T : Deposition velocity of aerosol due to thermophoresis (m/s)
v_P : Overall deposition velocity of aerosol (m/s)
x : Distance (m)

α : Plugging coefficient for laminar flow (m^2/kg)
β : Plugging coefficient for turbulent flow (m^4/kg^2)
γ : Ratio of heat capacity (−)
ε : Wall roughness of duct (m)
ε_g : Radiative emissivity from hot fluid to the wall (−)
ε_f : Radiative emissivity from flame to the wall (−)
η : Filtration efficiency or deposition efficiency (−)
ρ : Density of fluid (kg/m3)
σ : Stefan−Boltzmann coefficient ($J/m^2 sK^4$)

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Table I Major conditions and results of solvent fire tests for the flow rate of 30 air-volume changes per hours

Run number	Stages of Mixer settlers	Burning area (m^2)	Air-volume change rates (times/h)	Volumes of solvent and water phases (m^3)	Mass loss rates of solvent (kg/h)	Burning rates of solvent (kg/h)	Boilover burning times (s)
BOIL20A1	3	0.2304	30	0.03/0.03	3.6-5.3	3.3-4.7	4100
BOIL3001	6	0.4608	30	0.06/0.06	10.0-12.0	5.0-6.5	2790
BOIL3002	1	0.0768	30	0.01/0.01	0.94-1.2	0.91-1.2	4230
BOIL3003	9	0.6912	30	0.09/0.09	14.0-20.0	11.0-15.0	2500

Table II Major conditions of rocket fuel burning tests

Run number	Weights of rocket fuel (kg)	Burning times (s)	Length of second duct (m)	Ventilation flow rates (times/h)	Number of HEPA filters	Inserted pipe	Coments
EXPL20A1	2	5	50	30	12	none	
EXPL2002	4	5	50	6	12	none	
EXPL2003	4	5	10	6	12	none	
EXPL3001	6	5	50	6	10	none	* Observation of HEPA filter by the video
EXPL3002	6	5	50	6	12	none	
EXPL3004	10	5	50	6	12	none	
EXPL5002	6	5	50	6	12	yes	* 90mmϕx1000mm pipe

Table III Major conditions of red-oil explosion tests for 30 vol.%TBP/n-dodecane with nitric acid

Run number	Volumes of solvent (ℓ)	Volumes of nitric acid (ℓ)	Conc. of nitric acid (mol/ℓ)	Heater powers (kW)
NITR2004	1.0	1.0	7.0	0.85
NITR2011	1.0	1.0	10.0	0.85
NITR3001	1.5	0.5	13.2	0.85
NITR3002	1.9	0.1	13.1	3.50
NITR30A2	1.9	0.1	13.1	1.70
NITR3003	1.5	0.5	13.0	0.85
NITR3004	1.9	0.1	13.0	0.85
NITR3005	1.0	1.0	13.0	0.85

Table IV Basic Eqiations of CELVA-1D

Continuous equation

$$\frac{\partial(\rho A)}{\partial t} + \frac{\partial(\rho A v)}{\partial x} = \dot{M} A \quad (1)$$

Motion equation

$$\frac{\partial(\rho v)}{\partial t} + \frac{\partial(\rho v^2)}{\partial x} = \frac{\partial(P+q)}{\partial x} + \frac{\partial F}{\partial x} \quad (2)$$

Energy equation

$$\frac{\partial\{\rho A(e+v^2/2)\}}{\partial t} + \frac{\partial\{\rho A(e+v^2/2)\}}{\partial x} = \frac{\partial\{Av(P+q)\}}{\partial x} + A\dot{Q} - \dot{q}_w \frac{\partial A_w}{\partial x} \quad (3)$$

State equation

$$P = (\gamma - 1)\rho C_v T \quad (4)$$

Mass transport equation

$$A\frac{\partial C}{\partial t} + \frac{\partial}{\partial x} AvC(1-\eta) = A\{\dot{S} - \dot{R}\} \quad (5)$$

Loading equation of smoke on the HEPA filter

$$\frac{dL_{o,k}}{dt} = Av\sum_k \eta_k C_k = Av\sum_k (1-1/DF_k)C_k \quad (6)$$

Table V Analytical Functions and Mathematical Models of CELVA-1D

Analytical functions	Mathematical models
Basic equations for thermofluid dynamics	* One-dimensional compressible equations of continuity, momentum, energy and state, and calculations of flow resistance for cells, ducts, dampers, HEPA filters and blowers in the network.
Transports of smoke and aerosol in the network with these deposition	* One-dimensional mass transport equations for aerosol in the network. * Deposition of aerosol and smoke containing radioactive materials onto the wall of cells and ducts.
Numerical procedure	* Node-junction method and control volume method with Newton's numerical solution for the implicit time integration in CELVA-1D.
Source term models during fire and explosion	* Mass and energy release rates during fire and explosion in the cell. * Smoke/aerosol generation rates with the lognormal particle size distribution. * Release rate of radioactive materials into cell atmosphere during fire and explosion.
Heat transfer models	* Heat transfer to the wall and heat radiation calculated by the band model with radiative emissivity of burning gases. * One-dimensional partial differential equation of heat conduction into the wall of cells and ducts.
HEPA filter models	* Plugging equations of aerosol/smoke during fire and explosion in the ventilation system. * Filtration efficiency by the Kirsch filtration model.
Leakage model	* Release of radioactive materials to environment.

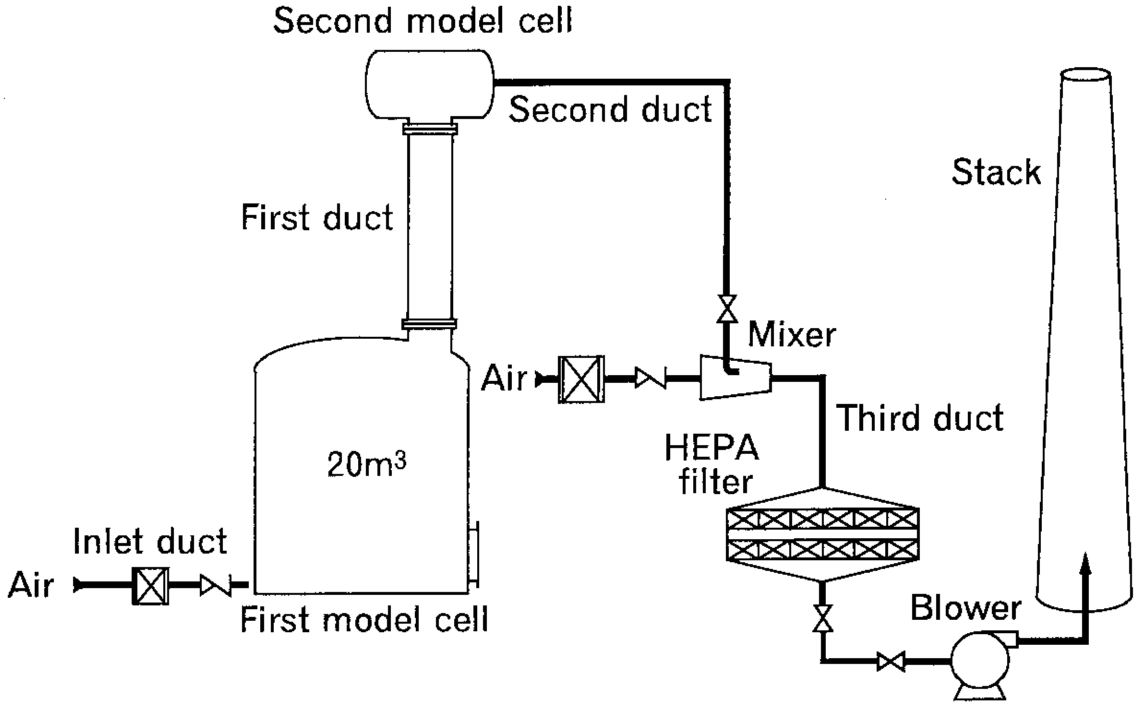


Fig. 1 Schematic of large-scale test facility

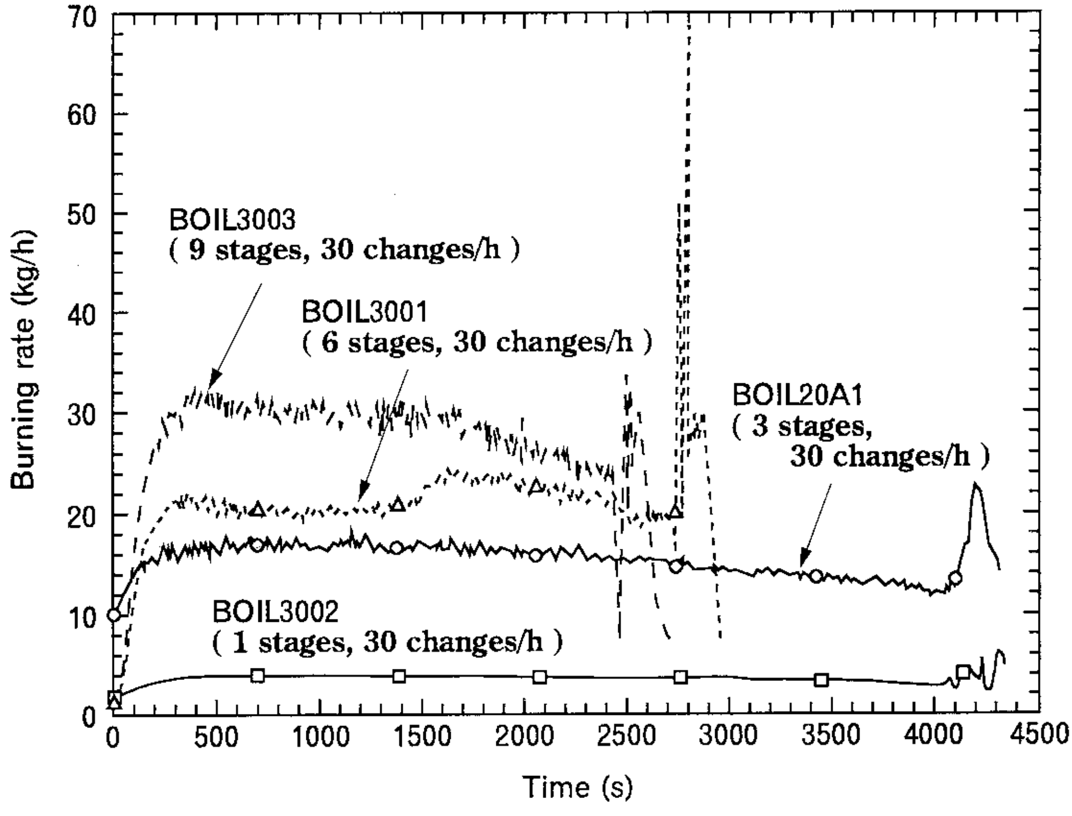


Fig.2 Burning rates of 30vol.%TBP−n−dodecane over times

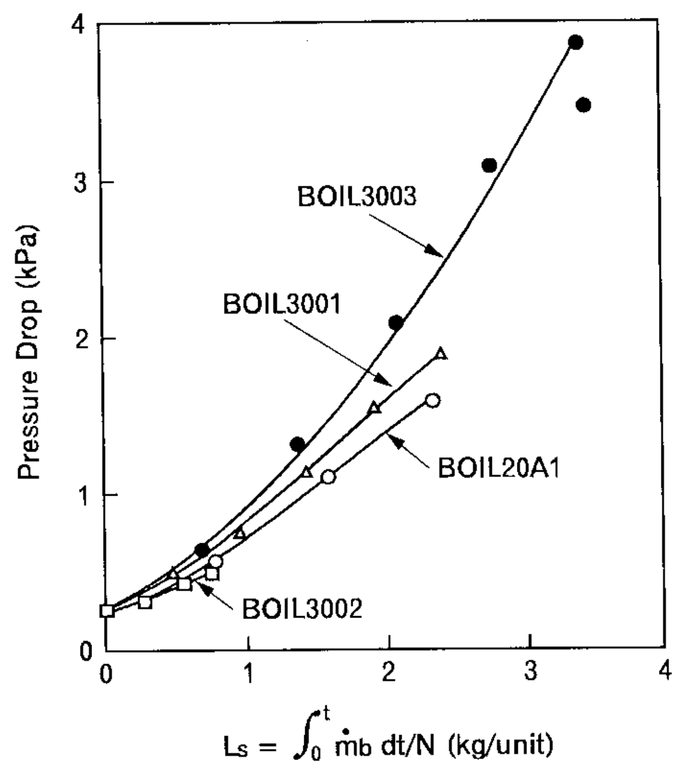


Fig.3 Pressure drop across the front stage of HEPA filters compared with cumulative mass weight of burned solvent per one unit of half–size filter

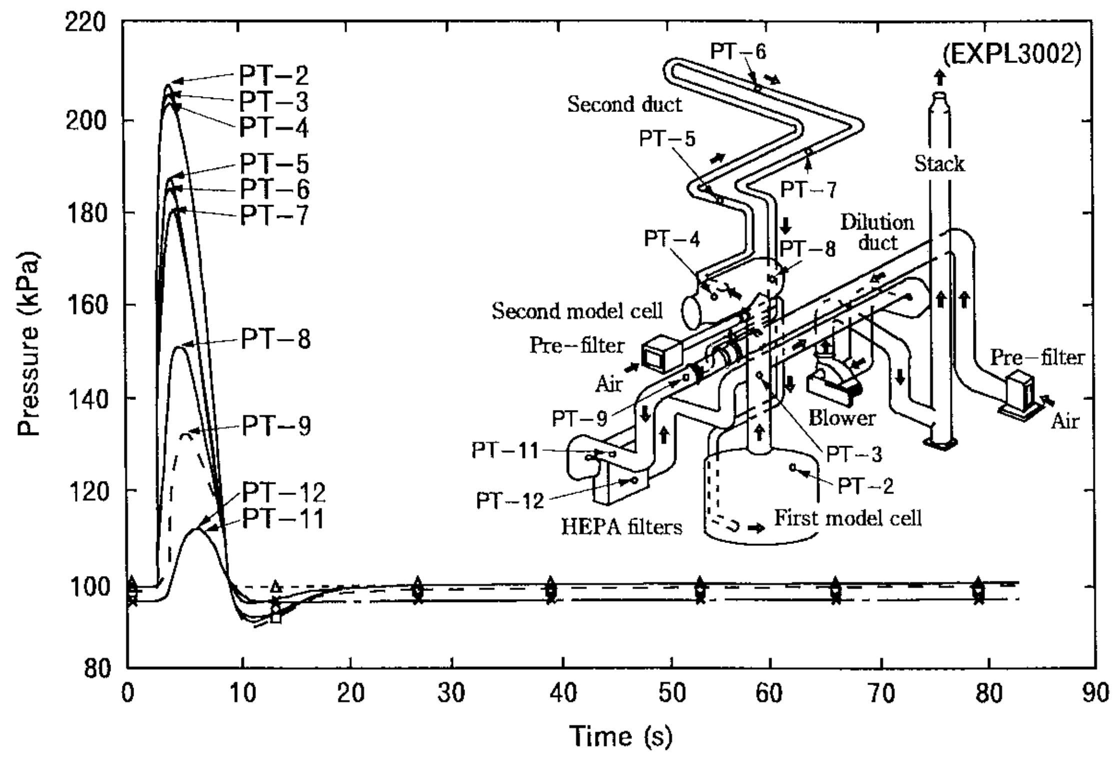


Fig.4 Propagation of pressures in the ventilation system of the test facility

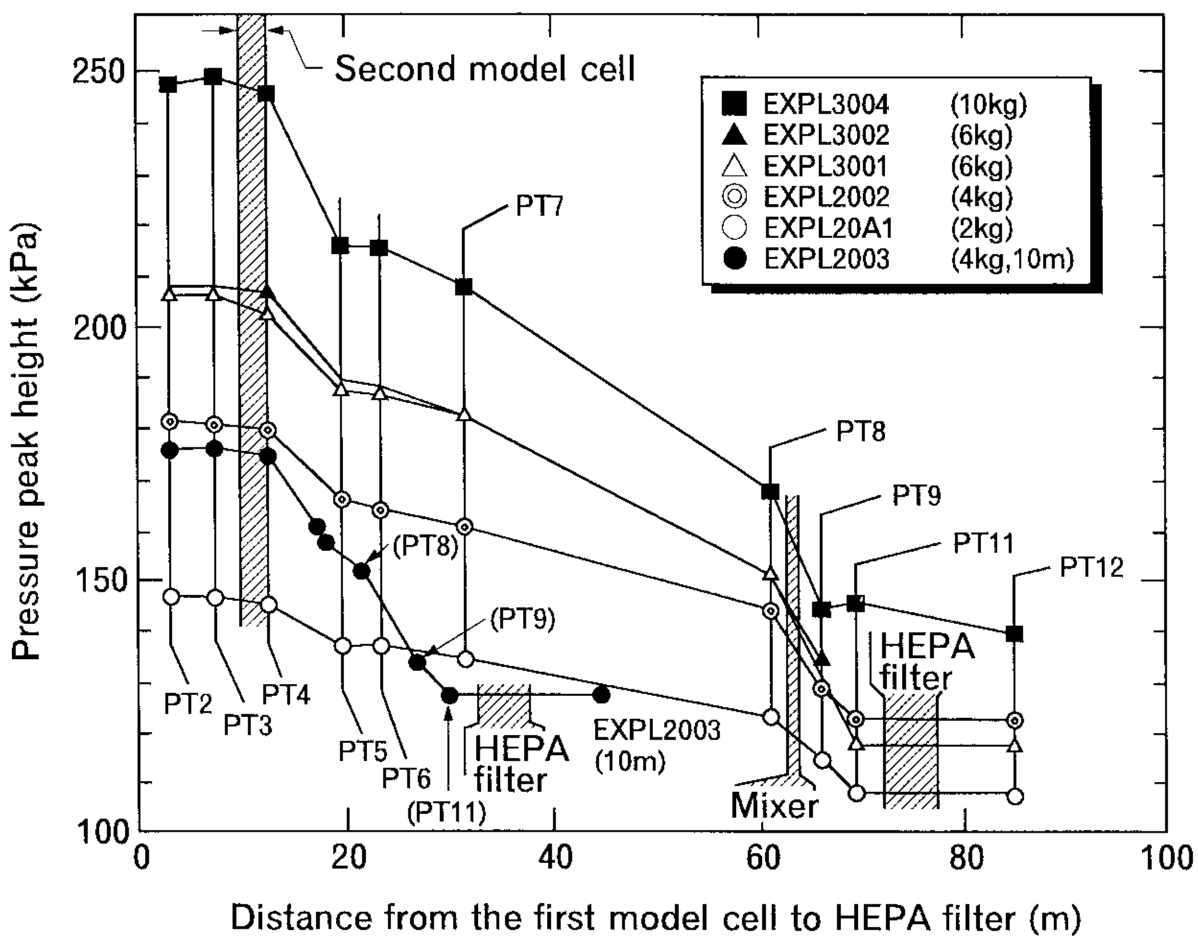


Fig.5 Decline of pressure peaks by the burning of rocket fuels from 2 to 10kg. The second duct was 50m long except in EXPL2003, where it was 10m long. Pressures were measured by sensors of the facility shown in Fig.4.

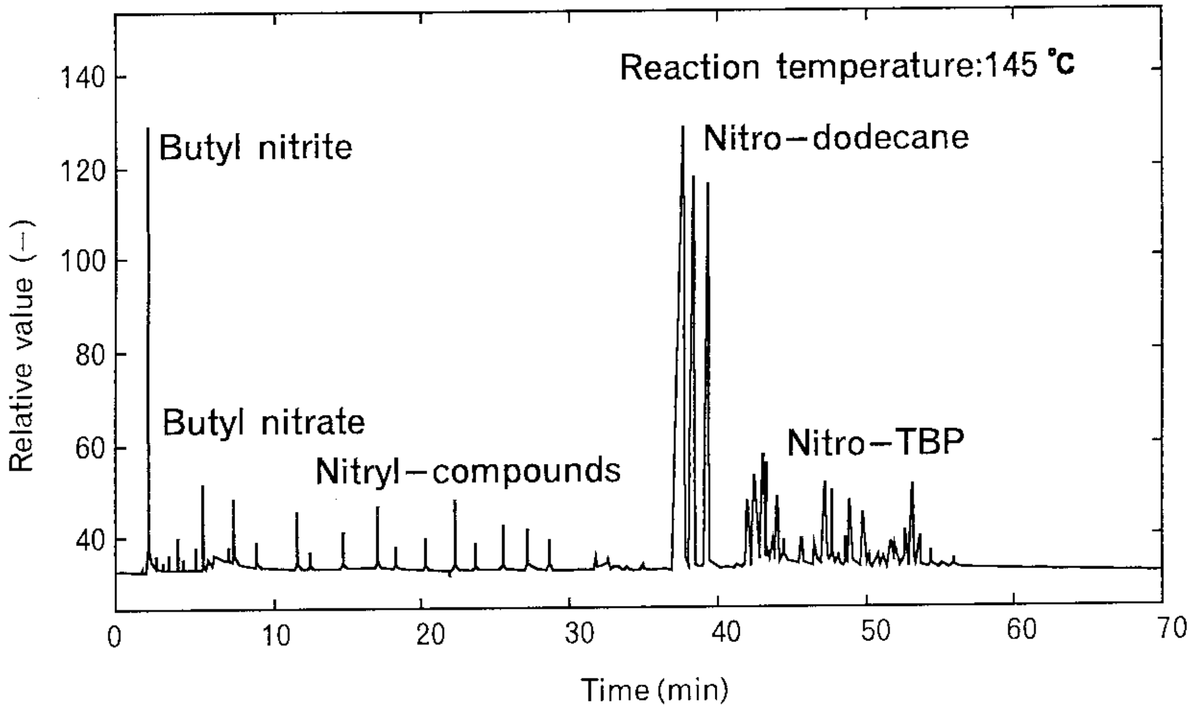


Fig.6 Compounds of red−oil in mixed 30vol.%TBP/n−dodecane solvent formed by heating 145℃

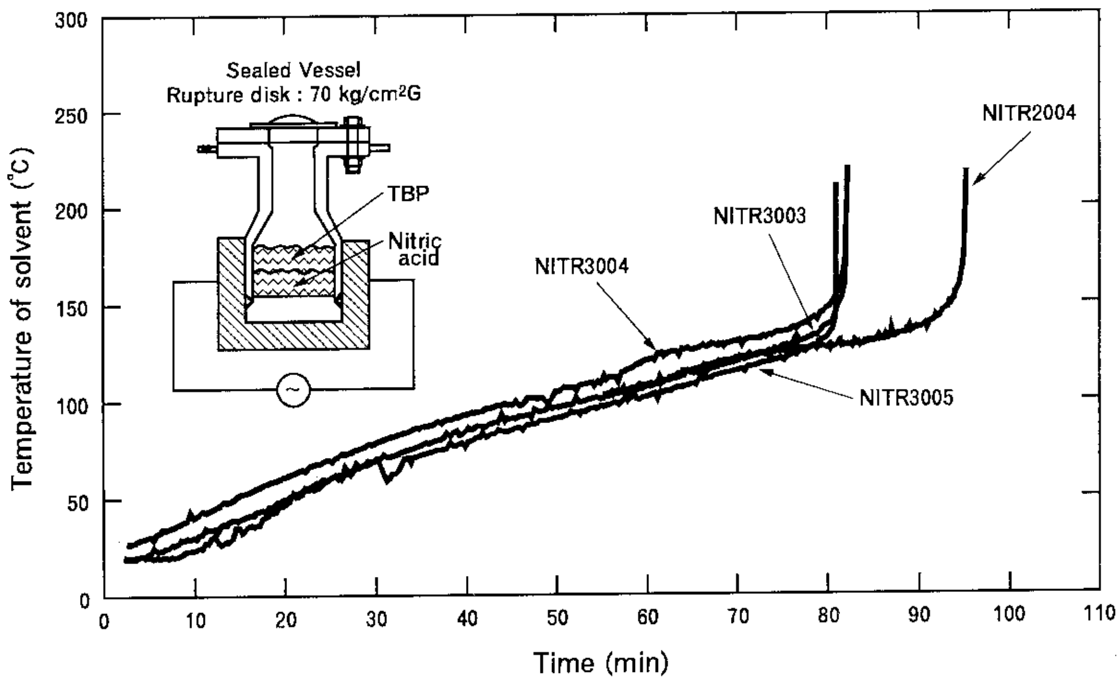


Fig.7 Temperature increase of solvent by the exothermic reaction for TBP with nitric acid in the sealed vessel

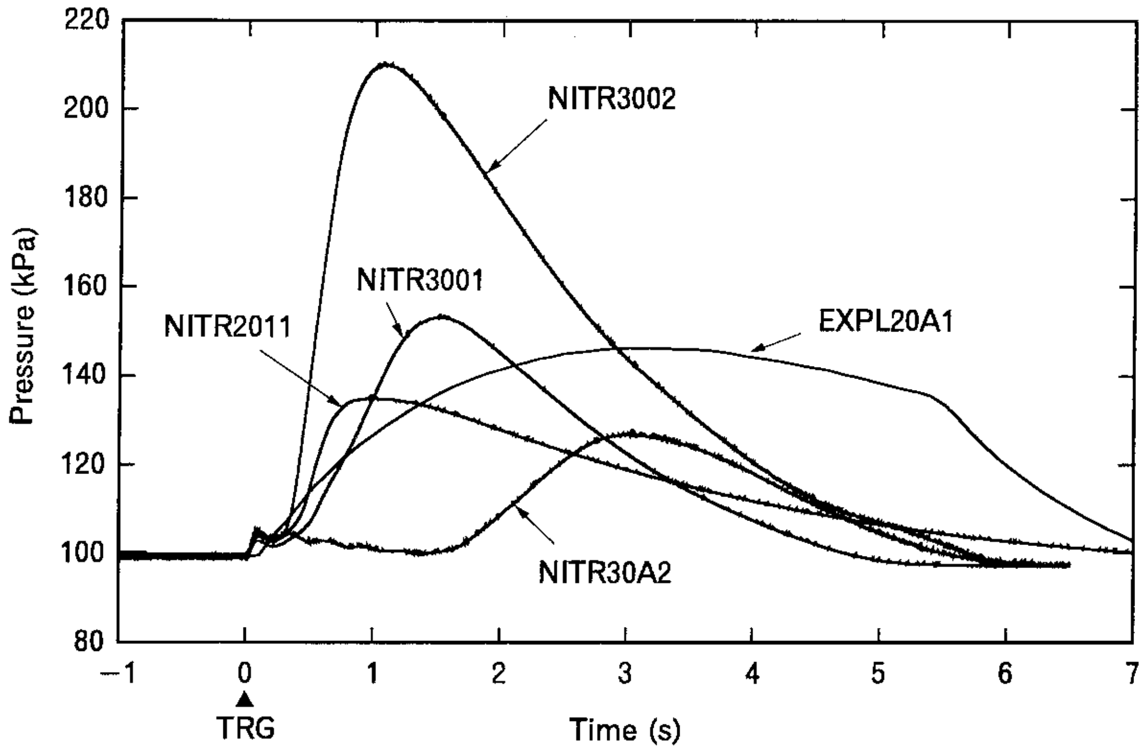


Fig.8 Pressure in the first model cell by red−oil explosions and the rocket fuel burning (EXPL20A1)

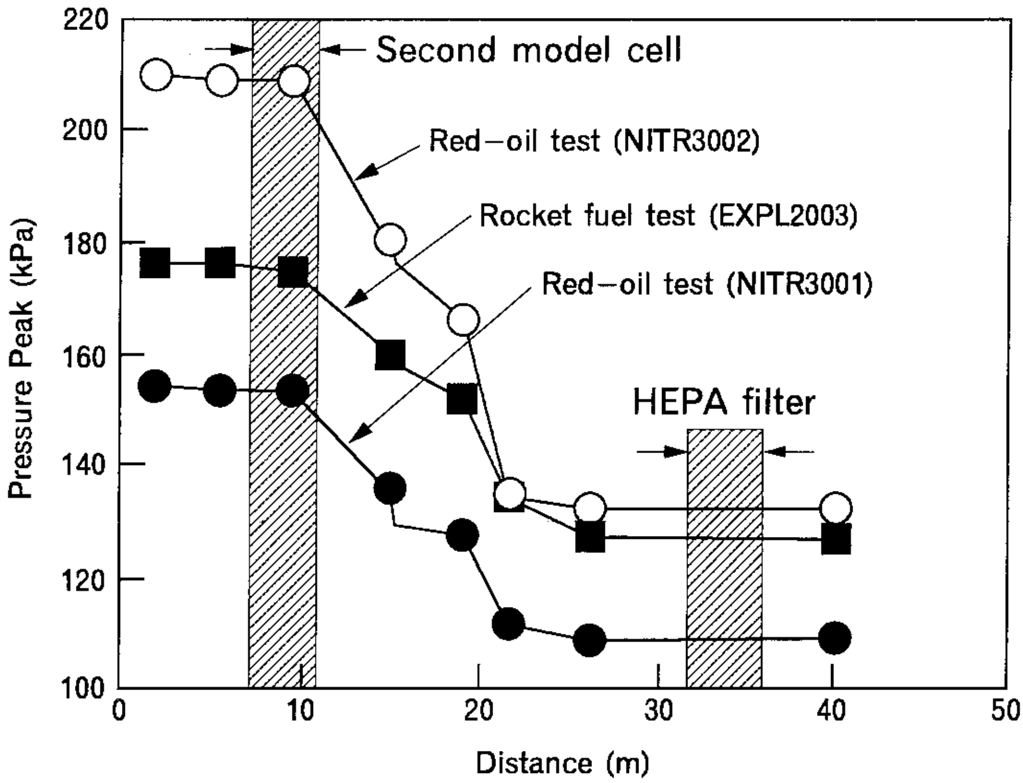
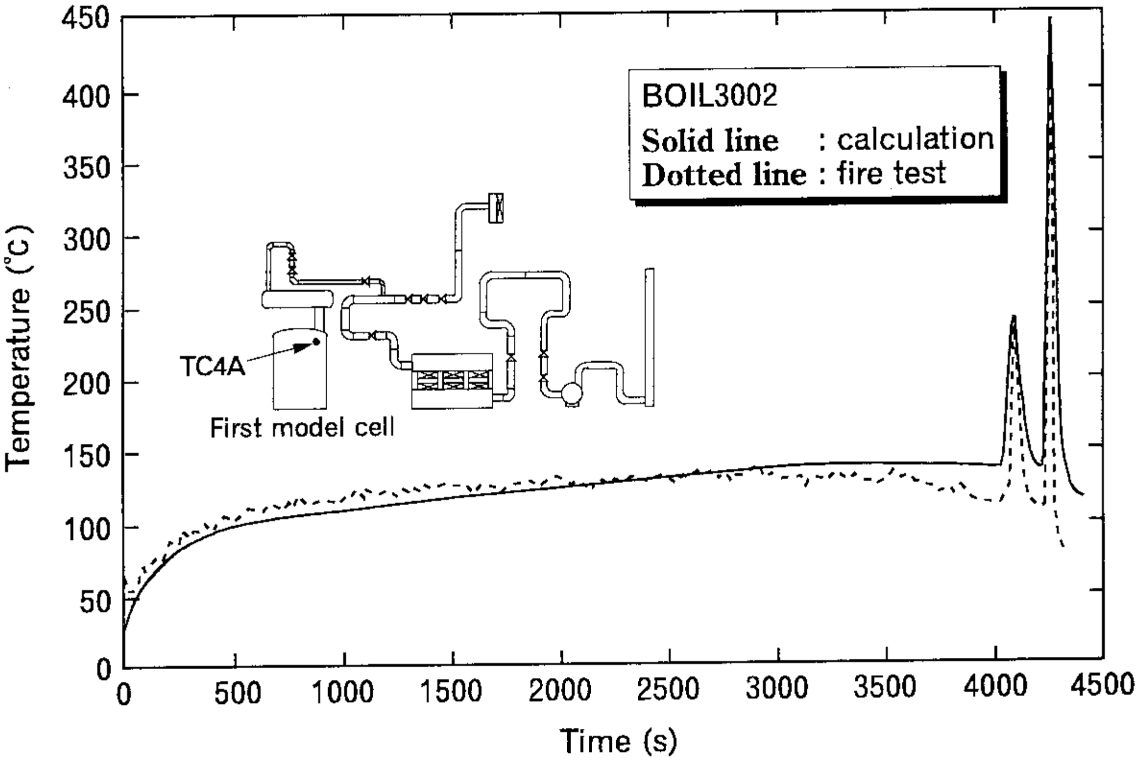
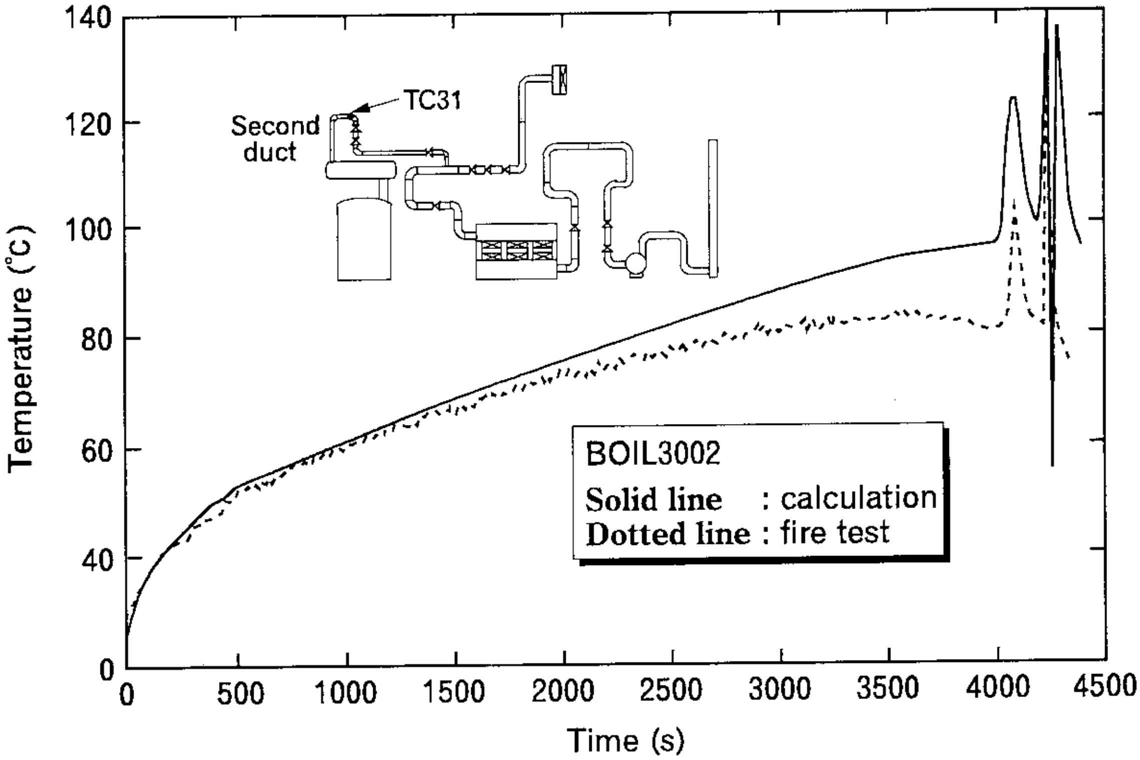


Fig.9 Decline of pressure peaks propagating through the facility



Tempareture in the first model cell vs. time



Tempareture in the second duct vs. time

Fig.10 Comparison of temperatures between code calculations and the measurements of fire test by large−scale test facility

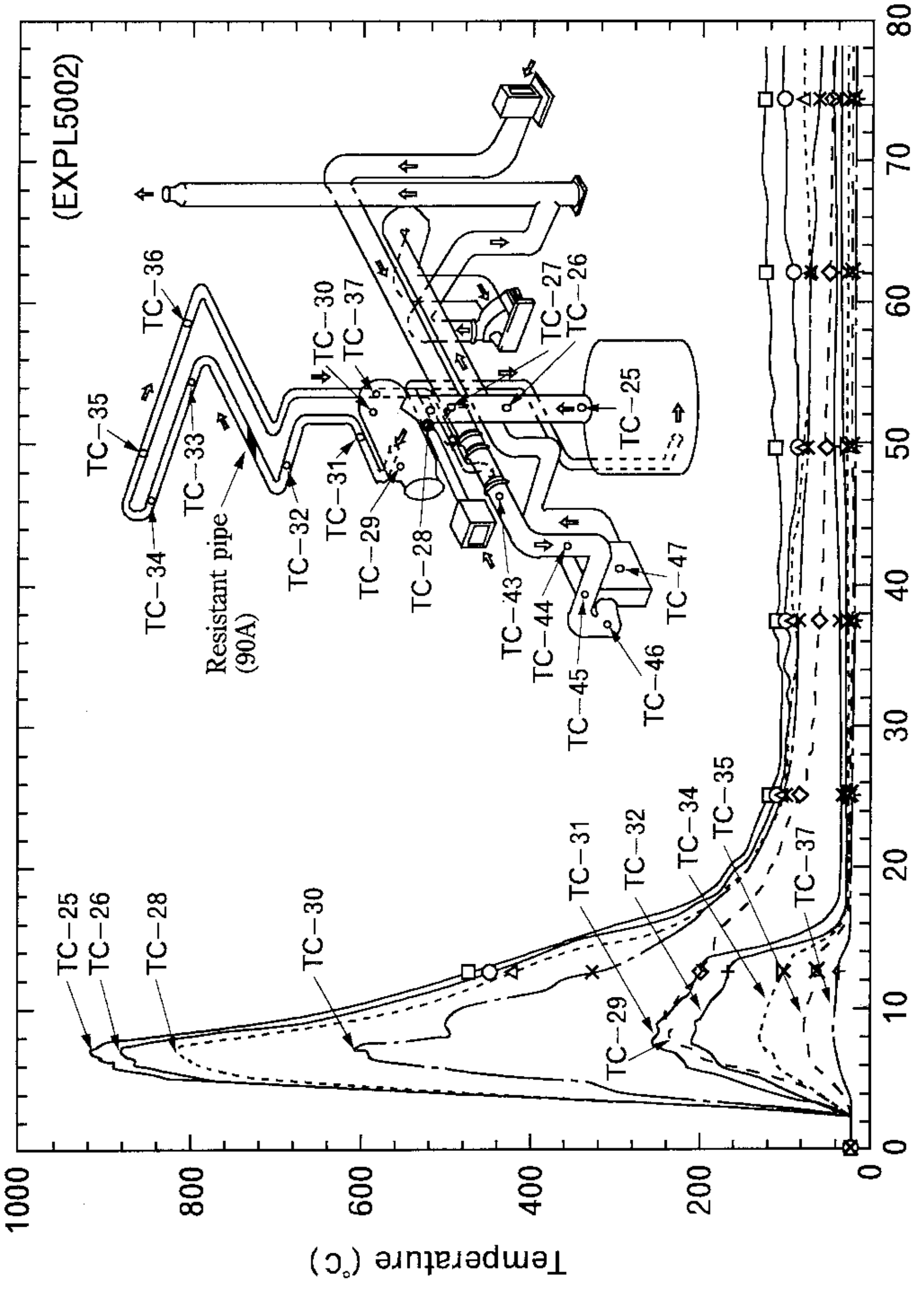
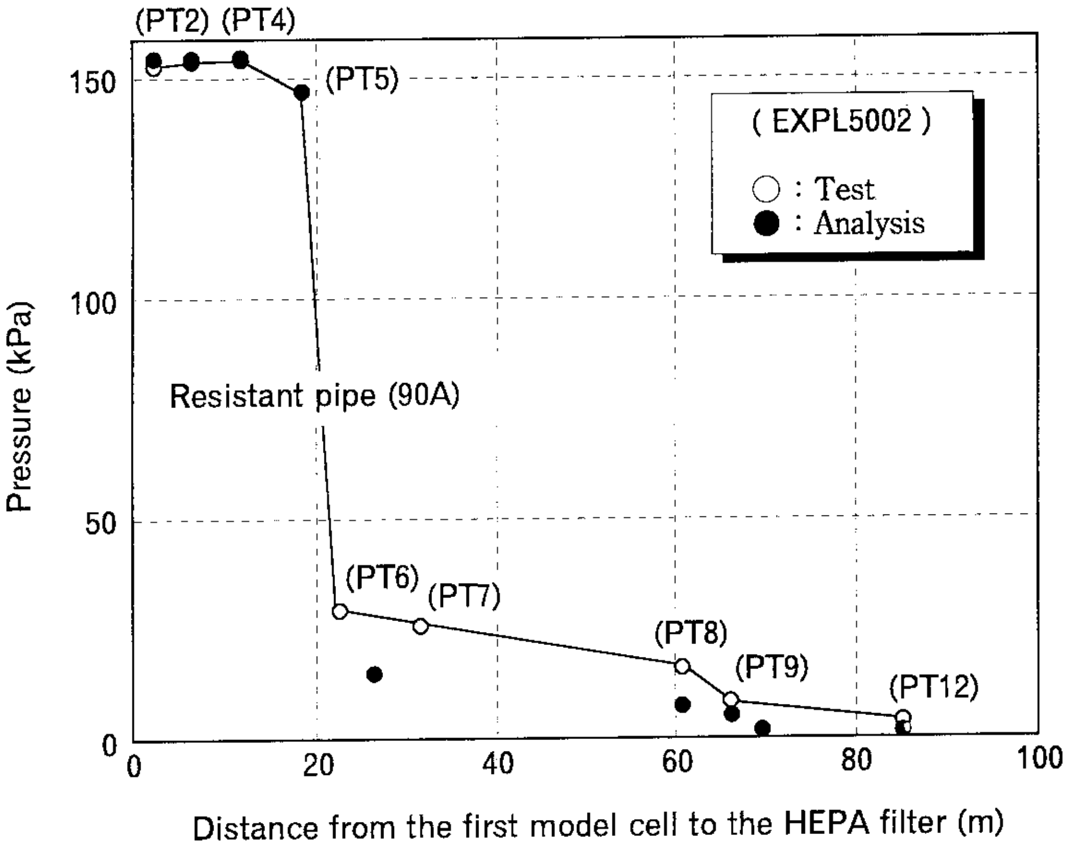


Fig.11 Propagation of temperatures in the ventilation system



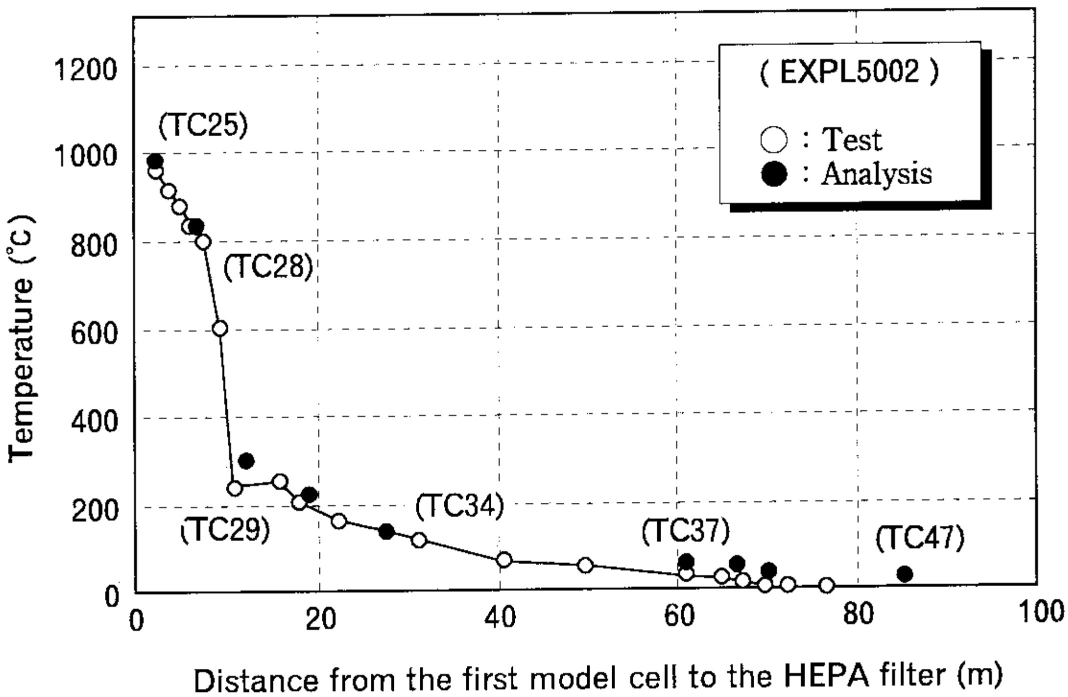


Fig.12 Comparison of propagated pressure and temperature peaks between code calculations and test results

INVESTIGATIONS CONCERNING FIRE-INDUCED ACCIDENTS IN NUCLEAR FACILITIES

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ABSTRACT

In the context of fire protection in technical buildings of French nuclear facilities, three principles have been adopted: prevention, detection and fire-fighting. Their implementation makes it possible on the one hand to limit the fire ignition and the fire growth, and on the other hand to prevent fire extent which would lead to unavailability of several safety related equipment.

Although progress has been made in this direction, the fire risks have still not been eliminated. It is therefore essential to evaluate the fire effects and to assess their consequences.

To this end, three main R&D programs have been conducted into fires.

Part I sets out the fire PSA methodology used for a 900 MWe PWR.

Part II gives an outline of two fire and ventilation computer codes useful for the fire PSA.

Finally, part III gives an outline of the tests already performed and those currently under way in the two laboratories of the Institut de Protection et de Sûreté Nucléaire (IPSN) in order to qualify the codes and provide useful information for the safety assessment.

PART I: METHODOLOGY OF THE FIRE PROBABILISTIC SAFETY ASSESSMENTS OF 900 MWE PRESSURIZED WATER REACTORS

1.1 Introduction

Fire Probabilistic Safety Assessment –called fire PSA– is being carried out by the Institut de Protection et de Sûreté Nucléaire (IPSN). The goal is to identify fire-induced core meltdown sequences and to calculate the corresponding probabilities.

This study, which considers the internal aggression resulting from fire occurring in various locations, is the continuation of the level 1 Probabilistic Safety Assessment–called PSA 900– already performed by the IPSN for the same French standardized series of nuclear power plants, taking into account internal events likely to occur in the installation (such as random failures of equipment or human errors). The overall probability of core meltdown obtained by PSA 900 is around 5.0 E-5 per reactor-year (refer to [1]).

The general objective of the fire PSA is to supplement the deterministic analyses on which the reactor design and the fire provisions and protection features are based in order to get better appraisal of nuclear risks related to fire and prioritization in efforts to reduce the risks. In particular, fire PSAs should make it possible to highlight the strong points and the weak points of the installations and of their operation in their various operating states, as regards the protective measures taken against fire hazards.

The fire PSA will notably include useful information for safety analyses such as, for example, the compartments in which the contribution of a fire is the most important for the probability of core meltdown.

A specific methodology has been developed for the fire PSA of unit 1 of Le Blayais nuclear power plant which has been chosen as the reference plant for the French series of 900 MWe PWR. Nevertheless its implementation is not immediate because much information is needed in order to qualify and/or quantify fire phenomena and fire-induced core meltdown sequences.

Consequently, an operational program is being elaborated in order to obtain the information needed for the PSA.

1.2 Methodology of the fire PSA

The following main tasks are being performed.

1.2.1 Inventory of data derived from experience feedback

The objective is to draw lessons from French PWR experience feedback of fire events by developing statistics as regards:

- the fire occurrence frequencies,
- the failure of the installed equipment, directly or indirectly (e.g. failure due to fire protection system release) subjected to a fire, and of the communicating elements between compartments (e.g. fire dampers and fire doors),
- the failure of fire detection and fire suppression systems,
- the time taken by the plant personnel and/or fire brigade to extinguish a fire.

Most of the information will come from Electricité De France fire event files.

1.2.2 Inventory of plant data

The objective is, on the one hand to locate fire initiators and the safety-related equipment, and on the other hand to collect the data necessary for the numerical simulation of the development of a fire in a compartment and its possible extent to adjacent compartments.

Three sources of information are useful: plant digital database, drawings and plant walkdowns.

The technical buildings are divided into zones –called fire zones– which include one or several compartments each. For the inventories, plant data checklists are established for information collection concerning each compartment in order to support the methodology implementation.

The following identifications are done zone by zone.

1.2.2.1 Location of equipment

All the equipment (mechanical, electrical, cables, detectors, extinguishers), safety or not safety-related, is listed, including its function and position designated by XYZ coordinates in each compartment. Particularly, cable tray numbers including type and voltage, cable wraps, equipment on both ends of cable (e.g. electrical cabinet, pump) are listed.

1.2.2.2 Combustibles and ignition sources

The equipment which may be involved in a combustion or at the origin of a fire is identified according to the two following criteria:

- transient and fixed combustibles:
    * combustible materials used during maintenance operations (e.g. wood, plastic products, waste, scraps, rags, flammable liquids),
    * lubricating oils for pumps,
    * diesel engine oil,
    * materials used for cable sheets (e.g. PVC), hydrogen,
    * etc.
- ignition sources:
    * mechanical equipment: large and small pumps, diesel generators, turbine generators, dryers, ventilation subsystems, elevator motors, air compressors,

* electrical equipment: electrical cabinets, junction boxes, transformers, batteries,
* human errors: welding and cutting processes,
* miscellaneous: hydrogen tanks, hot spots, trash,
* etc.

1.2.2.3 Fire barriers and communicating elements between compartments identification and description

Fire barriers and communicating elements between compartments (doors, dampers, ventilation ducts, etc.) are listed. Their possible failure (e.g. non closing, ageing) is subject to the study of fire extent between compartments.

Moreover, one needs the dimensions of the compartments, the dimensions of the communicating elements and other data (such as ventilation rates) to perform the fire numerical simulation study.

1.2.3 Fire occurrence frequencies

The fire initiators, identified in the experience feedback analysis, will be grouped into families to assess the corresponding fire occurrence frequencies.

Moreover, the inventory of equipment makes it possible to identify the compartments containing one or several fire initiators belonging to these families. It is thus possible to assess, for an initiator in a compartment, the single unit occurrence frequency of a fire. This assessment is carried out for all the initiators and all the compartments in which they are to be found.

1.2.4 Selection of critical zones

The number of zones which are subject to data inventory is considerable. These zones are selected and only those where a fire is likely to develop and to jeopardize safety are kept. The selected zones are called critical zones.

The main data needed to perform this stage are:

- compartment dimensions,
- concerning fixed and/or transient ignition sources: positions in compartment (XYZ coordinates), actual heat release rate, actual heat of combustion,
- concerning safety equipment: position in compartment (XYZ coordinates), thermal inertia, damage criteria (critical temperature and/or critical heat flux),
- automatic detection and suppression systems actuation time,
- availability of automatic suppression systems,
- probability of manual suppression effectiveness,
- availability of redundant safe shutdown systems.

The FIVE methodology (refer to [2]), an NRC-approved quantitative screening technique for fire analysis and critical zone selection, is used.

1.2.5 Fire scenarios

1.2.5.1 Identification of fire scenarios

Fire risk quantification requires the delineation of a large number of possible events.

A fire scenario is a chain of subsequent events. The first event is the fire initiation and the others– called « generic events »– generally involve the following possibilities:

- fire detection: plant personnel detects the fire, automatic detectors actuate, control room gets indirect indication (the early detection and late detection are considered separately),
- fire extinguishing: fire self-extinguishes by lack of combustible or of oxygen, automatic suppression system actuates, plant personnel/fire brigade uses portable extinguishers and/or hose reels,
- fire propagation: fire spreads to overhead cable, transient combustible, adjacent compartments due to lack (or delay) of intervention and/or failure of communicating elements.

In case of fire extent to an adjacent compartment which may be located in another critical zone, it is necessary to extend the fire scenarios, taking into account the « generic events » in the new compartment.

The scenarios including these possibilities are developed for all the critical zones and are represented by event trees –called fire event trees.

1.2.5.2 Studies of fire development and fire consequences

The objective is to predict more accurately by fire numerical simulations in the critical zones how long it would take to damage the equipment subjected to fire effects.

The FLAMME_S fire computer code (refer to part II) developed by the IPSN will be used.

A digital code database is created including characteristic data of common combustibles used in nuclear installations in order to aid the user.

It should be noted that the equipment time to damage will be calculated by fire computer codes using zone models which are not, in some particular cases, representative of the physical phenomena. Therefore, the simulation results cannot be used without an expert advice.

Moreover, the analysis of repercussions on actuators due to the damage of instrumentation and control cables requires particular precautions.

1.2.5.3 Probabilistic quantification of fire scenarios

The objective is, on the one hand to identify fire scenarios including destruction of equipment or fire extent to adjacent compartments, and on the other hand to assess equipment destruction probabilities.

The fire scenarios are modeled by fire event trees. Quantifying these scenarios requires the assessment of the generic event probabilities. Depending on the nature of the « generic event », this assessment will be carried out either directly from experience feedback or by using the fault tree method.

Failure of safety-related equipment due to automatic fire suppression systems or human errors made during operations specific to fire (e.g. fire suppression) will be also analyzed and quantified. Its assessment will be based as far as possible on the specific information derived from experience feedback.

Once each fire scenario is quantified, each target damage probability is assessed.

The main data needed to perform this stage are:

- automatic detection and suppression systems reliability,
- fire barrier reliability (fire door, dampers, etc.),
- human factors such as manual detection and suppression effectiveness.

The list of potential events and incident sequences must be as complete as possible. The events and sequences must be properly described.

1.2.6 Quantification of core meltdown sequences

The final work consists in assessing the probability of core meltdown sequences which can result from a fire. It includes quantifying:

- fault trees of systems equipment of which is aggressed by fire,
- event trees to assess the probability of core meltdown taking into account notably human factors such as operator response to ensure safe shutdown (e.g. error of diagnosis, spurious actuation).

The fault and event trees of PSA 900 level 1 will be used with proper adaptation if needed.

The RISK SPECTRUM (refer to [3]) software will be used to quantify fire-induced core meltdown sequences.

1.3. Conclusion

The above mentioned methodology is gradually tested to assess its applicability to PWR fire PSA and to find out all the elements necessary to perform each task.

Many data (e.g. equipment random failure probability) come from the PSA 900. Other elements come from bibliographic inquires and R & D investigations in the IPSN. In particular, many tests are performed in two laboratories of the IPSN in order to:

- determine the input parameters for digital fire simulation codes for various configurations and fuels used in nuclear installations;
- understand fire phenomena in order to make an expert judgment regarding the results of digital simulations.

It is advisable to note that carrying out the fire PSA represents a sizeable amount of work. It requires the assistance of several specialists in:

- fire safety science,
- PWR operation in incident and accident situations,
- statistic and probabilistic analyses,
- reactor operation,
- human factors.

PART II: FLAMME_S FIRE COMPUTER CODE AND SIMEVENT VENTILATION COMPUTER CODE

2.1 The FLAMME_S code

2.1.1 Purpose

The code is a complete reworking of earlier codes. It takes into account technical progress and improves the services offered for safety assessments.

It will be used particularly for the fire PSA of 900 MWe PWRs.

The code calculates the thermal effects and thermodynamic consequences –which are very useful for the safety assessments– of a fire in a compartment and those of its eventual extent to adjacent compartments through communicating elements.

2.1.2 General

The code is used to determine the fire duration and how the characteristic parameters associated with the development of the fire, such as pressure, temperature and composition of the different gases present in each compartment, vary with time.

A fire compartment contains a mixture of gases and combustion products. It is assumed to be non-deformable. The compartment may contain liquid or solid fuels. A compartment can also contain one or several cable trays as well as one or several pieces of equipment such as electrical cabinets.

Each compartment can be confined, and ventilated naturally or mechanically. The communicating elements for natural flow correspond to an orifice, a door or a damper whose surface can be a time function or a temperature function to simulate gradually the opening and closing operations.

The oxygen needed for combustion is taken from the zone in which the fire source is situated.

2.1.3 Simplifications adopted

These are the main hypotheses used when developing the code:

- the gases being inside a compartment can be described using a two-zones model or a single zone one,
- the fire plume can be described using the rZ coordinates (GUPTA model),
- the fuel, solid structures, and installed equipment are characterized by a mean temperature,
- the flame is a point source of radiant flux,
- the cable tray can ignite grid by grid,
- the gas zones and the flames are assumed to be semi-transparent,
- the walls and combustibles are assumed to be opaque, capable of diffusing heat,
- the static pressure is uniform in a compartment.

2.1.4 Mathematical modeling

Each system component (compartment, fuel, cable tray, installed equipment, wall, sprinkler) has its own set of discrete equations depending on the degree of physical representation desired:

- energy balance for a structure,
- conservation of energy and mass equations,
- mass and energy balances for each cumbustible,
- mass exchanges (e.g. plume, top and bottom zones, ventilation),
- heat exchanges (convection, radiation),
- combustion chemical reactions,
- etc.

2.1.5 Simulation capabilities

The code can be used to predict the following phenomena:

- temperatures of the gases (bottom zone, top zone, plume, in equipment), of the walls and of any equipment in the compartment,
- heat flux emitted by the flame and incident heat flux in walls and on installed equipment surface,
- oxygen and combustion products,
- pressure,
- mass rates of gas inflow and outflow via openings,
- dysfunction and time to damage of equipment,
- etc.

2.1.6 Quality assurance

The code is subject to a specific quality assurance procedure. This quality assurance procedure requires, in particular, the production of a « life history » for the code, known as the « qualification dossier » containing the nine main following documents:

1) Functional specifications (design principles of the code).
2) Technical specifications (detail of the design).
3) Physical and mathematical models (formulae used in the development phase).
4) Numerical descriptions (approximations of the formulae used).
5) Computer description (code and its computer environment).
6) Qualification report (comparison of tests and calculations).

N.B. At present, more than 20 representative tests for fire in nuclear facilities are used for the qualification.

7) Data sensitivity study (data uncertainties and numerical sensitivity).

8) User's guide (adequate data selection).

9) User's manual including guide examples.

Furthermore, modifications or regrades of the code lead to draw up a new version. Computer tests and updates are then carried out on any new qualification dossier before the new version of the code enters service.

2.1.7 Conclusion

Version A of the code is operational. Qualification work for a new version (comparison between tests and calculations) is currently under way, especially for options involving fires with several ignition sources, fires in several compartments, propagation of combustion to vertical or horizontal solid fuels, GUPTA plume model.

A test program has been drawn up to meet the qualification needs in order of priority.

2.2 The SIMEVENT code

The ventilation code, SIMEVENT, was developed by the IPSN, COGEMA and SGN.

2.2.1 Purpose

The code is used to model the operation of a ventilation network, and to simulate incidents which could affect it in order to assess the safety consequences.

2.2.2 General

In a nuclear fuel reprocessing plant, a fire may increase greatly the quantity of radioactive materials carried through the ventilation system. Therefore, it is necessary to assess the transfer of radioactive materials from the fire compartment inside the nuclear installation to the point at which they are released into the environment, passing via several containment barriers.

This code is developed to be applied from design until dismantling for any nuclear installation, both under normal and some abnormal conditions (e.g. fire).

The operating principle behind the SIMEVENT code is based on dividing the ventilation system into nodes which are connected by branches.

2.2.3 Simplifications adopted

The flow is assumed to be one-dimensional, non-viscous and incompressible.

2.2.4 Mathematical modeling

The complexity of physical phenomena such as fluid motion in the ventilation system requires some simplifications of the actual situation. The following simplifications are adopted in the code:

- balance at a node: a node represents a volume where the pressure and temperature are considered as uniform for each case studied. In addition, the mixture of matters is assumed to be homogeneous and there is no deposition at the node. The conservation of mass equation is used.
- balance in a branch: a branch represents a segment of the circuit surrounded by two nodes and fully determined, in terms of ventilation, by a relationship of the type $\Delta P = f(Q)$. Bernoulli's, momentum and material transport equations are used.

It should be noted that the models for deposition in conduits and filters, according to the particle size distribution, are being carried out.

2.2.5 Simulation capabilities

In its current version, the code can calculate the pressure, mass flow rates and transport of material in the ventilation system. The different disruptions dealt with are:

- variable positions of a valve, a leak and an orifice,
- shutdown of a ventilator,

- closure of a damper and/or opening of a valve,
- variations of pressure and temperature caused by fire.

2.2.6 Conclusion

Studies are currently underway to update the code, with better consideration being given to heat dissipation, transport and deposition phenomena in the ventilation system. Moreover, the aerosols and combustion products can plug the filters of the ventilation system. This phenomenon is currently being studied. Furthermore, extending the code to fires implies combining it with a code which can describe the development of the fire in a compartment and also the introduction of models to describe the effects associated with fire (e.g. heat loss in the conduits and deposition of materials).

On another hand, under fire conditions, the ventilation may remove heat, smoke and combustion products from the fire compartment. Moreover, in many ventilated compartments of the nuclear installation, the ventilation system will be stopped and the fire damper will be closed automatically or manually to confine the fire and the combustion products. Combustion products and/or radioactive materials should be released, if possible, after fire extinction: then oxygen is not brought in the compartment during the fire. Because of this, a study has begun concerning the opportunity and feasibility of combining SIMEVENT with FLAMME_S.

PART III: FIRE TESTS IN TWO IPSN LABORATORIES

3.1 Introduction

The IPSN has several laboratories. Two of them perform fire tests, one at the Grenoble research center, and the other at the Cadarache research center. Several chambers were built there (volumes ranging from 5 m^3 to 3600 m^3) for use in small, medium and large scale tests in various configurations.

The tests conducted have the following two main objectives:

- determining input parameters for fire and ventilation computer codes for various configurations and fuels commonly used in nuclear installations,
- obtaining a better understanding of fire phenomena in order to carry out, especially, an expert judgment.

3.2 Experimental readings

The tests were performed in various ventilation conditions to measure the following main parameters:

- burning: rate of burning, propagation of the fire at the surface of the fuel, chemistry of the phenomenon, actual heat release rate and heat of combustion,
- flame: shape, size and temperature,
- combustion products: concentrations of gases and aerosols,
- plume: temperature and entrainment phenomena,
- temperature and pressure of the gases in the chambers,
- temperature and flow rate of the gases through communicating elements (with mechanical or natural ventilation),
- temperature and heat flux at the walls of the targets and of the chamber,
- pressure and flow rate of the gases in each node of the ventilation.

In addition, transport of matters and deposition phenomena in a ventilation system are under study.

The test result analysis will notably allow the code database to be supplemented.

3.3 Tests already performed

Some forty large-scale tests were performed in various configurations with fuels commonly used in nuclear installations. The main fuels used are:

- TPH and a mixture of TBP and TPH (solvents used in reprocessing plants),
- DTE medium oil: a fuel mainly used in PWR pumps,
- PMMA: a material used in nuclear laboratories,
- PVC: a material used for cables and in nuclear laboratories,
- bitume: a material used in waste facilities,
- sodium: a metal used in FBR,
- cable trays.

3.4 Tests planned

A program of experiments, which comes under the IPSN's « five year plan », is currently underway in order to better understand the fire phenomena and particularly to qualify the new versions of the codes. The aim of the program is to increase the confidence given to the results of digital simulation. This program includes, more specifically:

- various fire source locations (center, against a wall, in the corner of the compartment) and the orientations (horizontal or vertical for solid fuel) of the fuel,
- fire propagation from the first ignition source to other combustible materials located in the same compartment,
- fire propagation from one compartment to one or several other compartments through communicating elements,
- intermediate and full scale cable tray fire.

For information, two large-scale tests will be performed before the end of 1995.

The first will be performed in order to study the plume phenomena and thermal events which could cause failures and/or combustion of cable trays and cabinets. Five energized cable trays will be installed in the plume, below the ceiling out of the plume and above the floor near the ignition source. The cabinet, which will not be energized, will be placed on the floor, also near the ignition source.

The second, also full scale, is intended to study the dysfunction and combustion of cables in cable trays. For this test, twelve cable trays will be used.

3.5 Conclusion

The tests performed have, in particular, made it possible to qualify the codes.

All the comparisons of test and calculation values show the sensitivity of the code, especially to parameters associated with the ignition source and with gas flow phenomena. Research is going on with the development of formulae and suitable tests.

CONCLUSION

A large research and development program is being carried out at IPSN to evaluate the risks and consequences of fire induced accidents. Full scale experiments are conducted. The formulation and computer codes are being improved in order to cover more and more situations closer and closer to reality. The fire PSA methodology is sharpened to supplement the deterministic studies by taking into account the fire safety consequence analysis. Moreover, safety regulatories are constantly improved. Technical progress is also made to minimize fire ignition and fire effects which might jeopardize safety.

Therefore, fire phenomena in nuclear installations should be better controlled.

References

[1] **IPSN** - *A Probabilistic Safety Assessment of the standard French 900 MWe Pressurized Water Reactors (EPS 900),* Main report, April 1990.

[2] *Fire-Induced Vulnerability Evaluation (FIVE),* EPRI TR - 100370, Project 3000 - 41, Final report, April 1992.

[3] *RISK SPECTRUM version 2.1,* RELCON TEKNIK AB, BOX 1288 S-172 25, Sundbyberg, SWEDEN, April 1994.

PROBABILISTIC RISK ASSESSMENT FOR BACK-END FACILITIES: IMPROVING THE TREATMENT OF FIRE & EXPLOSION SCENARIOS

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The nuclear reprocessing facilities at Sellafield are a key component of the International business of BNFL. The operations carried out at the site extend from the receipt and storage of irradiated fuel, chemical reprocessing, plutonium and uranium finishing, through mixed oxide fuel production. Additionally there are a wide range of supporting processes including solid waste encapsulation, vitrification, liquid waste evaporation and treatment. Decommissioning of the site's older facilities is also proceeding. The comprehensive range of these activities requires that the safety assessment team keeps up to date with developments in the field, as well as conducting and sponsoring appropriate research into methodologies and modelling in order to deliver a cost effective, timely service.

This paper will review the role of Probabilistic Risk Assessment (PRA) in safety cases for operations at Sellafield and go on to describe some areas of PRA methodology development in the UK and in which BNFL is a contributor. Finally the paper will summarise some specific areas of methodology development associated with improving the modelling of fire and explosion hazards which are specific to BNFL.

SAFETY ASSESSMENT AND THE ROLE OF PRA

The basis of a complete and robust safety case lies in a comprehensive hazard identification process. BNFL employs the Hazard and Operability Study approach (HAZOP) (Ref 1) and has developed specialised sets of key words specifically tailored to facilitate the examination of nuclear chemical processes (Fig 1). The use of computer based systems to prompt and record the study has been introduced and is undergoing refinement. The potential initiating events for hazards are then grouped into similar, related topics to facilitate the production of hazard assessment reports, each addressing a logical grouping - for example loss of containment scenarios. Fig 2 illustrates the components of a typical hazard assessment.

Criteria

The demonstration of safety using a PRA based methodology requires accident risk criteria to be defined. Such criteria have been in use at Sellafield since the early 1980s and consequently in their present form (Ref 2) they benefit from over a decade of development and refinement. Most specifically they take cognisance of the two major publications issued by the UK Regulators (Ref 3 and 4) which define the standards against which the acceptability of BNFL's licensing submissions will be judged.

The primary criteria are concerned with the acceptability of risk from the whole of the Sellafield site from fault conditions with allocations of that total risk being assigned to individual plants. The fundamental principle is that the summed risk of death from operations on site to members of the public should be less than 10^{-6} per year and that the summed risk of death for a worker on a plant should not exceed 10^{-5} per year. In deriving criteria, a mortality risk factor is used which relates the risk of death to the amount of low level radiation received. In the analysis of risk to members of the public, appropriate critical groups have been identified. These critical groups, represent people who because of either their location, employment or lifestyle represent the most exposed members of the public. Due consideration is given to both aerial and marine discharges, and appropriate critical groups are identified for each type of release. For example the critical group for an aerial release will be those individuals who live or work close to the site boundary whereas the group identified as most at risk from a marine discharge could be for example, fishermen working the estuary or people living on boats.

In addition to the primary criteria based on this principle, ie those which limit the summed mortality risk for the most exposed member of the relevant group, a series of secondary criteria have been developed. These additional criteria account for the sensitivity of the public to maloperations on the plant and in particular consequence aversion. They limit the frequencies of certain categories of events to levels below that which would be necessary on pure risk grounds.

For each criteria there is, therefore, a consequence and frequency component. For example, the summed frequency of an operator receiving an accidental radiation exposure in excess of 50 mSv must not be greater than 1E-3 y^{-1}. Clearly the consequence threshold must be breached in order for a frequency target to be applicable.

Finally there are low levels of consequence which can best be regarded as trivial. The criteria, therefore include a cut off point below which no further assessment is required.

Modelling Consequences

When modelling a particular scenario, the normal technique is to identify wherever possible a 'bounding case' which adequately covers the identified fault or fault set. This is done by modelling the scenario for the worst relevant case as appropriate.

For example, if the analysis concerns lifting a load, in order to assess damage caused by dropping the load, the heaviest load normally lifted is used. If, however, we are concerned with loss of containment following a drop the highest potential release is assessed. This may be either the flask which sustains the greatest damage on impact or the one which carries the greatest inventory, depending on which will bound the risk - which by implication includes frequency consideration.

The first stage of the assessment is therefore to calculate the potential consequences of the fault. These are calculated, where appropriate, for workers in the plant that is affected, workers in the surrounding areas and for relevant members of the public. Clearly these populations will vary with the fault being studied and the type of release mechanism. Coincident failures of cell extract and gas clean-up systems are considered as a route to such exposures.

A set of databases are maintained which provide input to the analyst's calculations. These include a consequence database containing release fractions and decontamination factors which has been built up over a considerable number of years. The information is gathered from both published sources and sponsored experimental work and is backed up by plant history and event experience from Sellafield, together with other events reported around the world.

Where an event can be shown, even on the basis of grossly pessimistic assumptions, to result in negligible or trivial consequences it will not be necessary to refine the assessment further. For example, for a public aerial release, if the estimated committed dose to a member of the public can be shown to be less than 10 μSv (all pathways) no further assessment need be carried out. To put this figure in context the average annual dose to members of the public in Britain from natural radioactivity is about 2200 μSv (with areas as high as 6000 μSv). Another example is that at the typical cruising height of a passenger aircraft, a dose of around 10 μSv would result from a two hour flight.

Analysing Frequencies

Where a frequency calculation is required, standard techniques for constructing fault trees are used. A computer code has been developed specifically for this purpose. Common cause failures are accounted for within that code using our own model. For more complex scenarios complementary methodology is available to allow the use of event trees.

Frequency calculations require specific information on plant equipment performance, which is obtained from a reliability database. For operational plant, reliability data is derived from analysing the actual performance of equipment in service. Over the years BNFL have invested heavily in gathering such data. Sellafield staff have collected information on approximately 2000 equipment groups, comprising some 30,000 individual items and over 50,000 individual faults. In the case of a new plant at the design stage information is extracted from a variety of sources including manufacturers' literature and published databases. Prior to the use of such data a unique data sheet is generated by reliability engineers, who provide advice on usage and applicability of all data. In addition, they check values and failure modes against experience in other plant with similar operating conditions to ensure consistency. Similarly a 'human factors' database is available, with experts providing advice and ultimately independent verification.

Each item of information extracted from the databases for use in assessments is uniquely referenced and traceable. This facilitates the Quality Assurance checks undertaken on HAZANs by suitably experienced personnel within the department, both for accuracy and applicability.

Robustness of the Safety Case

Having completed the consequence calculations and any required frequency estimation the results are then compared with the relevant criteria. Some events will not have breached a consequence threshold, whilst others will have been subject to a full consequence and frequency analysis. This comparison will show which events, if any, breach or approach criteria. While it is extremely rare for events to approach the primary risk criteria, it is less so for secondary criteria.

The final stage of the assessment requires a study of the sensitivity of the estimated risk to possible variations in the basic event data to be performed whenever the primary risk target is approached within an order of magnitude by a single fault. This study addresses both consequences and frequencies. The sensitivity of the estimated consequence is investigated to explore the effects of data variations and establish the particular elements of the model to which the result is sensitive; this allows the assessor to examine the model for excessive pessimism, or to provide more robust supporting information when the safety case requires it.

Generic Approaches

As with any technique which requires expert judgement to construct a model of the real situation, there is a tendency for variations in the approach taken to the issue and in presentational style. Where similar fault scenarios exist on a range of plants, these individual variations in approach not only incur a cost penalty at the assessment stage as each individual assessor seeks his own approach to the issue, but carry the additional potential overheads of increased costs during the QA stages of assessment validation, confusion amongst plant operators and challenges from the regulator.

Thus the potential exists for benefit to be gained from standardised approaches to a range of commonly encountered hazards. At Sellafield we have sought to capitalise on these gains by producing a range of guidance notes which specify preferred approaches to assessors.

PRA DEVELOPMENT IN THE UK

In this section the current research and development activities at the national level in the United Kingdom are described. The programme is supported by BNFL and the nuclear power utilities, Nuclear Electric and Scottish Nuclear. The UK Regulator, the Nuclear Installations Inspectorate plays a significant role in the definition and monitoring of the programme.

To aid the description, current activities can be grouped under two broad headings, validation of PRA and PRA methodology development.

Validation of PRA

Looking at validation first, it is perhaps not surprising to find a scoping study to aid definition of the range of validation required by examining the extent of uses to which PRA is put. For each type of use the specific objectives will be listed in order to identify the key validation requirements.

A project on 'fault tree consistency' is considered necessary because constructing fault trees requires 'judgements' on the part of the assessor judgement on what aspects are significant and should be included in the model and how to set up the model. In the project, different assessors will be asked to construct a fault tree based on a specified problem and with identical reliability databases. The fault trees will then be compared, and a judgement will be made on how significant any differences are.

'Accident sequence precursor analysis' notes that some fault trees predict some plant abnormalities at high frequencies eg 1 y^{-1}. The project seeks to investigate how predictions compare with reality.

A project on 'reliability data' has been set up to answer the question "should reliability data be collected for components or for systems?". Fortunately, in this area BNFL is well served by the results of its internal investment in data collection and analysis.

PRA Methodology

It is important to take proper account of 'common cause failure' (CCF) in PRAs. Many CCF methodologies exist, each having a claim to be the most accurate, or easiest to use. The project will set up an operational experience database for CCFs which should allow validation and calibration of methodologies.

The 'uncertain analysis' project is required because PRA results are a mathematical combination of many base events, for each of which the mean value has an uncertainty associated with it. It is far from clear what effect uncertainties in base event values has on the uncertainty associated with the end event values; this project will attempt to clarify this effect.

'Worker risk' is generally assessed by PRA on nuclear chemical plants (including Sellafield) but not on nuclear power plants. The project will query whether the same techniques are applicable to nuclear power plants, or whether any modifications will be necessary.

'Living PRA' uses an on-line computer model to allow a reactor operator to monitor the risk rate associated with current equipment outages. Hopefully, this is used to ensure that risk is kept low by avoiding 'bunching' of outages. BNFL do not see this as a significant advantage in chemical plants: since many associated hazards are intermittent, the instantaneous risk rate is inherently highly variable, and cannot sensibly be held within narrow bands. To use a simple analogy, the level of risk we find acceptable in a plane coming in to land we would not find acceptable if sustained for the entire duration of a flight from the UK to Japan. Our views on the tolerability of the risks associated with air travel are based on the balance between the overall risk and the overall benefit. Nevertheless the project will seek to explore extending the living PRA concept beyond the very few reactors in the UK where it is currently being applied.

The 'event based seismic PRA criteria' project begins by noting that seismic PRA criteria are currently based on risks which are acceptable to the public under 'normal' circumstances. However, in the event of a severe earthquake, the direct non-nuclear consequences may be so large as to make the nuclear risks trivial by comparison. The ambition of this project is to test the proposal that nuclear criteria might be based on the non-nuclear risks presented by each earthquake 'size'.

SPECIFIC BNFL DEVELOPMENTS

Having described above the general areas of methodology development in the UK, this final section will discuss some examples of where BNFL safety assessors have generated novel or enhanced approaches to benefit Sellafield plant operations.

An assessment might require detailed re-examination for a number of reasons.

Overpessimistic modelling of the consequence of a potential fault scenario, even where the frequency targets are comfortably met, can create difficulties. It can result in the overspecification of protective systems and put unnecessary constraints on the operating envelop of the plant. It can also have the effect of sensitising the regulator; regulators tend to have a natural aversion to high consequence events at any frequency. In turn this can lead to requirements to make plant modifications, with operators incurring dose uptake and funds being committed on a false premise.

Alternatively a plant operation which had been adequately covered by a relatively simple assessment, perhaps one of the standardised generic approaches discussed above, may be the subject of a proposal to modify the process which causes the assessment to breakdown.

In principle the route to improving an analysis is straightforward. Since the analysis is essentially a model of a real, complex situation it will normally contain a number of simplifying assumptions., there is usually scope to re-examine the modelling assumptions utilising plant specific details and specialist advice. The skill lies in knowing which aspects of the model to investigate to best effect the required improvement!

The re-examination process can proceed in several ways (Fig 3). The most common methods are to look first at reducing either the consequence or the assessed frequency of occurrence. However assessors should not lose sight of the option of reviewing the literature or commissioning further experimental work in order to demonstrate that the perceived hazard cannot arise. For example, either the physical performance of flask drop tests or the application of stress analysis codes might reveal that containment is not lost if a flask is dropped in a hoistwell.

It should also be borne in mind that a positive outcome to the re-examination process is not guaranteed. For example, the assessor may form the opinion that whilst the model is pessimistic, there is no justifiable or cost effective enhancement that can be made, at least in the short term.

Hydrogen from U Metal Fuel and Cladding

Prior to the commissioning of the Magnox Encapsulation Plant swarf, from the decanning of uranium metal fuel (Magnox) was sent for interim storage in water filled silos. Considerable quantities of this material continue to be stored awaiting retrieval and encapsulation. Corrosion processes create hydroxides and release hydrogen gas. Consequently an identified hazard is a hydrogen fire initiated by a failure of the plant's ventilation system, resulting in hydrogen build up and a flammable atmosphere being created. An ignition source is assumed to be present because it is difficult to demonstrate the complete absence of one. In order to reduce the frequency of this perceived hazard to an acceptable level, air ejectors were fitted to the plant as an additional means of controlling the build up of hydrogen should normal and back-up extract systems fail.

The re-examination of the safety case looked into the effect of hydrogen buoyancy. An analysis revealed that even in still atmospheric air, ie with no wind creating a draft effect in the stack, the buoyancy of the hydrogen enriched air was sufficient to keep the hydrogen levels within the plant below the flammable limit. An important ingredient in making this case was the data collected on hydrogen evolution rates.

The result of this re-examination is a realisation that even with a failure of the ventilation system, a dangerous accumulation of hydrogen gas will not occur on the plant. The air ejectors are not needed, avoiding a proposed $0.7M refurbishment programme.

There is an additional benefit. In the event of a ventilation failure and the use of air ejectors, the radioactive discharge of the plant would be increased. The new arrangements, while they would still produce a slight discharge over the plant norm (if called upon), would give a much lower radioactive discharge than the previous arrangements. Consequently a reduction in pessimism led to a safer arrangement.

Radiolytic Hydrogen

The chemical processes at Sellafield expose water to high radiation fields, both as a solvent in reprocessing streams and in subsequent treatment steps. Radiolysis takes place producing hydrogen and other gases, whose composition depends on the chemical nature of the radiolysed liquor. Accumulation and combustion of radiolytic gases has been recognised as having the potential to cause significant plant damage.

The existing approach essentially assumed the threat from hydrogen generation to be real by making very pessimistic assumptions concerning the rate of hydrogen generation, how it might accumulate in plant process vessels, the extent of damage done in any explosion and the consequent release of activity. The standard approach allowed almost every scenario which could not be dismissed immediately to become a high consequence hazard. The safety case was made by relying on active removal devices and achieving the frequency target.

However, there were one or two problems with this simplistic approach:

- it led to the identification of high consequence scenarios which attracted regulatory concern
- on old plant it gave devices which actively purged hydrogen (eg level indication pneumercators) a higher safety significance than designed for
- on new plant it led to considerable expense in the provision and maintenance of engineered purging systems
- it was increasingly seen as unreasonably pessimistic when compared with real plant data and engineering judgement.

Consequently, a more realistic approach has been developed. The simplistic approach outlined above has become an initial screening stage. Where it shows a potential problem, the matter is then considered more carefully to establish the possibility of radiolytic gas combustion, and if a significant release could result. The structure of the additional consideration takes the following form:

- for this system, what is known about actual rates of hydrogen production ? (system chemistry may mop-up hydrogen, geometry or lack of agitation may depress rates well below those reported for short-term small-scale lab work)
- what (active) adventitious or passive dispersal processes can be identified, and would those which can be quantified allow a flammable mixture to be retained?

- if so, how long would it take to reach flammable concentrations, and could other significant thresholds be exceeded (mixture gives significant pressure rise on combustion, mixture is detonable) ?
- what is the worst pressure rise or shock wave that could be expected, and would the containing vessels withstand them (if so, consequences will probably be much lower) ?
- where would failure be expected to take place (and hence what is the radioactive inventory participating in any release) ?

It may then be possible to show that no release will take place, or that it will be less severe or involve a lower inventory, or that only under exceptional circumstances (hence at a very low frequency) could a release take place.

Shearing of BWR Fuel with Water Channel in Place

The baseload and post baseload contracts for reprocessing oxide fuel in THORP require the removal of the external water channels, sometimes called shrouds, from BWR fuel. A number of utilities have requested BNFL to receive fuel with the water channels still in place. The objective is to reduce dose uptake to power station operators. Consequently BNFL re-examined the THORP safety case against the idea of shearing BWR fuel into the dissolver with the zircaloy water channel in place. This investigation revealed that the only significant issue related to the potential for increased zircaloy fines from the shearing process which could result in a dust cloud explosion. All other issues from increased fines, for example nitric acid reaction and ignition of dry fines were already within the bounds set by the current safety case.

A review of both Sellafield and ORNL shearing data was performed. This data indicated the main variables in zircaloy fines production. For example, hydriding of zircaloy increases brittleness and this in turn produces more fines. The total cross sectional area of zircalloy in an element is also a factor in the amount of fines produced on shearing. The examination of this work also allowed the identification of good simulants to be identified for both the behaviour of the UO_2 fuel and reactor grade zirconium.

The next step was to establish a rig to look more closely at the sensitivity of the production fines. The identification of the simulant has allowed this research to be performed in a timely and cost effective manner, avoiding the complications of handling uranium and the costs of reactor grade zircaloy. However the simulants have been validated against the actual materials. The research programme has investigated the effect of the number of pins in an element and the position of cut within an element eg a cut through the pin plenum, fuel region and grid position.

As a result of this R&D programme BNFL have shown that shrouded BWR fuel from the utilities requesting the change will not produce more zircaloy fines than a standard 15 x 15 PWR element with zircaloy clad fuel pins. Consequently, the shearing of such fuel will lie within the existing safety envelope.

Uranium Hydride

Uranium hydride forms as a corrosion product from the storage of uranium metal fuel under water. Uranium hydride is pyrophoric in concentrations above about 10% if allowed to dry and exposed to air. Consequently the potential for fire, initiated by hydride but spreading to a more general uranium fuel fire has been recognised in U metal fuel handling plant. A fuel fire would be an event with major off-site consequences.

During the normal operation of these plants the scenario has been shown to be acceptable by virtue of the predicted low frequency. The key elements of making the frequency argument were:

- prompt reprocessing which limits the extent of Magnox cladding corrosion, which in turn limits fuel corrosion
- limited amounts of fuel with swollen ends, which split the cladding and had high surface areas of uranium for the hydride to form
- keeping the fuel and swarf wet at all times

The need to clean up and decommission the older metal fuel facilities has prompted a more detailed examination of uranium hydride formation in order to derive a more realistic model for the safety case. Every step in the chain from fuel corrosion to the onset of a fuel fire has been re-examined.

Looking at the onset of uranium metal corrosion, the conditions must be unusual to result in significant hydride production. There must be no oxygen present. There must be a high hydrogen concentration. The hydride must not be exposed to liquid water. High hydride levels occur only following limited water ingress through small perforations in fuel cladding. Uranium corrosion with liquid water always results in non hazardous hydride levels in the uranium corrosion products.

In the event of extended fuel storage, as will be the case for fuel encountered in decommissioning operations, slow oxidation occurs, and the hydride is destroyed. Further production of hydride is mitigated against by the progressive corrosion of the fuel cladding and the more general exposure of uranium to water. It is observed that after some 3 years of storage, the corrosion of fuel with swollen ends has progressed to a point where the end region (which is the most susceptible area for hydride production and the initiation of a serious uranium fire) has been corroded away.

It has been calculated that sufficient hydride to ignite a fuel rod can exist only in swollen fuel rod ends with their high surface area and thin layers of uranium metal. Therefore the high consequences associated with a fuel fire can only occur with fuel with swollen end regions covered in high concentrations of hydride. There is a limited potential for small fires to occur from hydride protected beneath residual cladding. However for a hydride fire to occur dry hydride must be exposed to air. Storage and handling fuel rods in wet conditions, together with the low radiolytic heat released by old fuel rods means that fuel will take periods of several shifts to dry out completely, reducing the frequency of even minor fires.

The use of this information results in uranium hydride ceasing to be a major concern across a large range of typical decommissioning operations. It also allows the highlighting of specific areas where the potential for hydride to be present may be rather more significant, for example where fuel has been stored long term in PIE bottles.

CONCLUSIONS

Probabilistic Risk Assessment is a central component of the BNFL approach to safety assurance. It is well suited to the wide range of processes carried out on the Sellafield site. The company was one of the pioneers in the introduction of the methodology to the nuclear chemical industry developing numerical criteria against which the acceptability of operations could be judged. Now, two decades later, BNFL continues to invest in the development of the technique.

With the maturity of the methodology assessors are adopting standardised, generic approaches to improve consistency and lower assessment costs. This allows assessment effort to be invested in re-examining high consequence or operation limiting scenarios and the benefits of four such re-examinations have been discussed in this paper.

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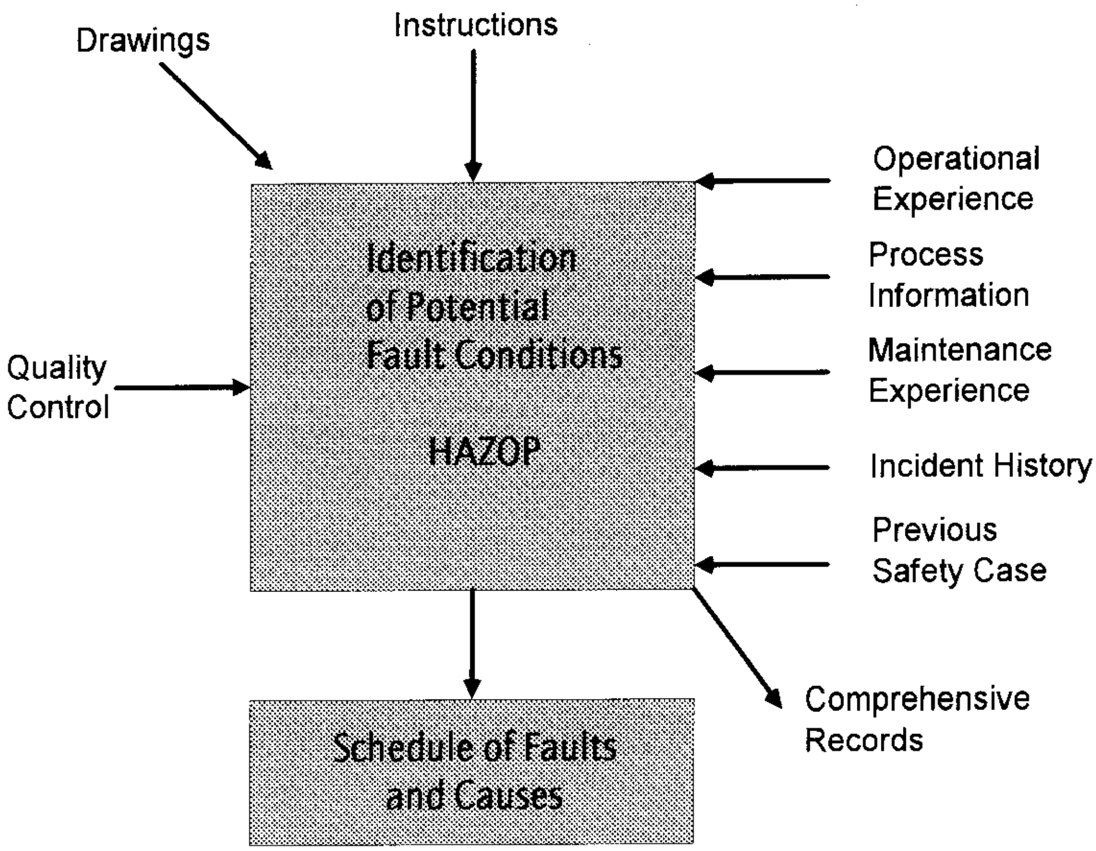

Fig 1 - The HAZOP Process

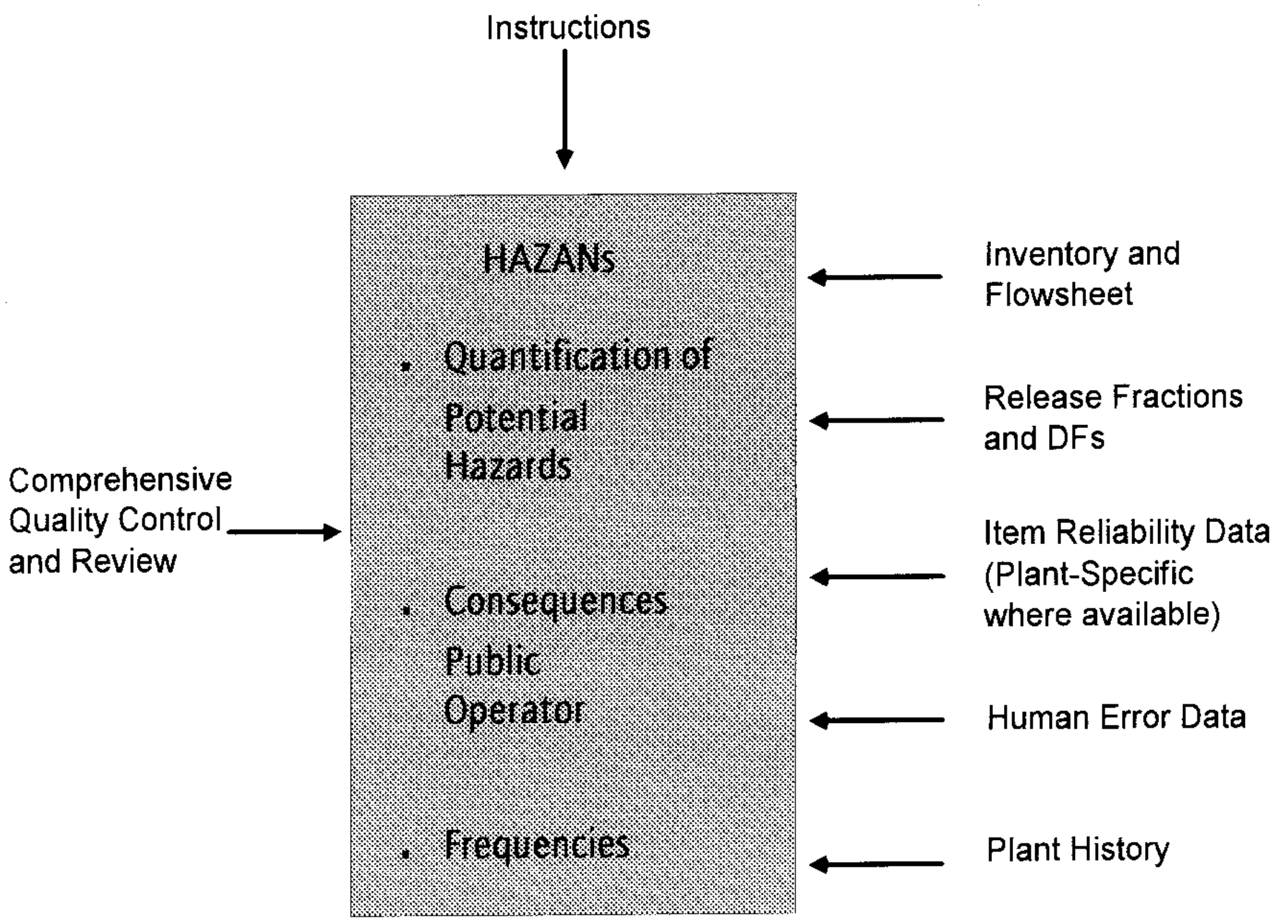

Fig 2 - Inputs to a Hazard Analysis

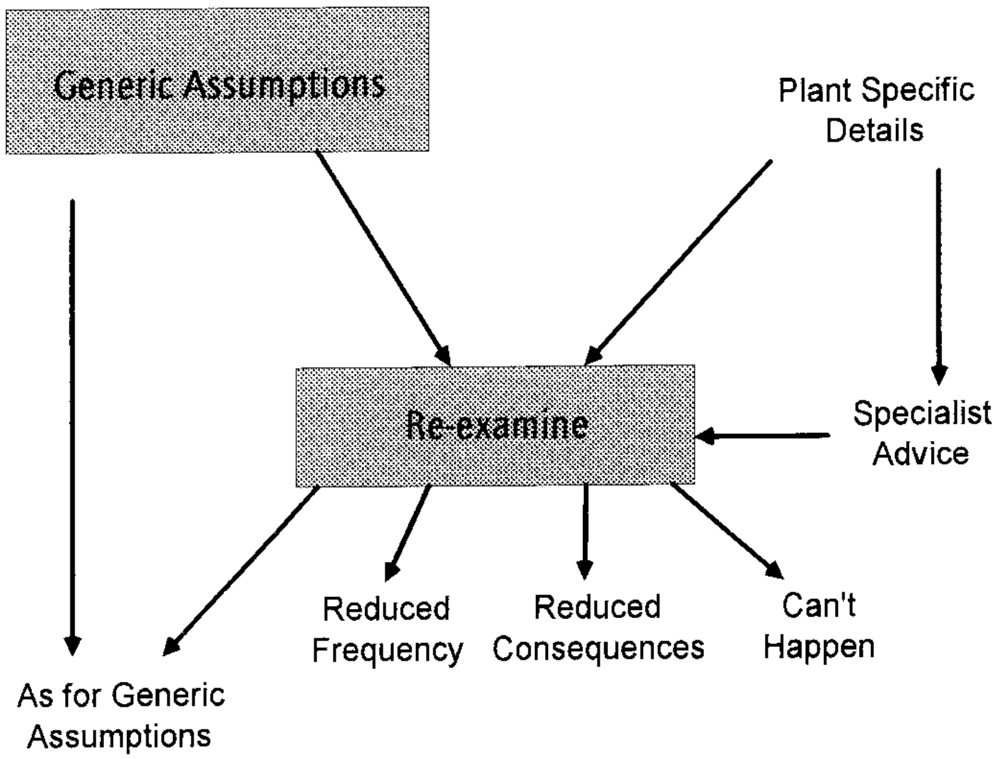

Fig 3 - Re-examining Safety Assessment

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Research on Reduction of Radioactivity Release and Database

Cochairs

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AERIAL AND LIQUID EFFLUENT TREATMENT IN BNFL'S THERMAL OXIDE REPROCESSING PLANT (THORP)

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Abstract

British Nuclear Fuels plc (BNFL) completed construction of its Thermal Oxide Reprocessing Plant (THORP) at Sellafield in 1992, at a cost of £1,850M. After Government and Regulatory approval, active commissioning was initiated in January 1994. Since then, the whole of the plant has been progressively commissioned and moved towards full operational status.

From the outset, the need to protect the workforce, the public and the environment in general from the plant's discharges was clearly recognised. The design intent was to limit radiation exposure of members of the general public to " As Low as Reasonably Practicable" (ALARP). Furthermore no member of the most highly exposed (critical) group should receive an annual dose exceeding 50 microsieverts from either the aerial or marine discharge routes.

This paper describes how the design intent has been met, concentrating mainly on aerial discharges. It describes the sub-division of the plant's ventilation system into a number of separate systems, according to the volume and source of the arising and the complexity of the treatment process. The dissolver off-gas, central off-gas, cell and building ventilation systems are described, together with the development programme which was undertaken to address the more demanding aspects of the performance specification. This ranged from small-scale experiments with irradiated fuel to inactive pilot plant trials and full-scale plant measurements. In addition wind tunnel tests were employed to assist dispersion modelling of the gases as they are discharged from the THORP stack. All the resulting information was then used, with the aid of mathematical models, in the design of an off-gas treatment system which could achieve the overall goal.

As far as liquid discharges are concerned, the paper describes the overall philosophy behind the THORP flowsheets which allows the environmental impact target to be met. The treatment of some waste streams is undertaken in dedicated equipment within the THORP building. Other streams are mixed with effluents from elsewhere on the site and treated in existing site facilities. After treatment and sentencing the effluents containing low levels of radioactivity are discharged down a 3 km pipeline into the Irish Sea.

Taken together, the aerial and liquid discharges from THORP are predicted to fall well within the original design intent of 50μSv. Aerial discharges are predicted to give a maximum critical group dose of 22μSv and marine discharges will give a dose of 36μSv to a separate critical group. Moreover the discharges are also predicted to fall comfortably within the limits on individual radionuclides which have been imposed as part of the Government's Aerial and Liquid Discharge Authorisations.

The detailed attention given to the development and design phases of the effluent treatment system should ensure that these predictions are mirrored by actual performance as the plant becomes fully operational.

INTRODUCTION

Brief History

THORP is the third reprocessing plant to be constructed on BNFL's Sellafield site in Cumbria, UK. The plant is designed to reprocess irradiated fuel from Advanced Gas-Cooled Reactors (AGR) in the UK and from Light Water Reactors (LWR) in Europe and Japan: virtually all of its first 10 years operation are committed with contracts for 7000t(U).

Conceptual design of the plant started in 1974, supported by construction and operation of a number of major development facilities during the later 1970's and early 1980's. Design of the plant was delayed by the Government's decision to hold a public inquiry (the Windscale Inquiry) in 1977. Following the successful outcome of the inquiry, design recommenced in 1983, leading to the start of civil construction in 1984 and eventual handover of the completed plant in 1992. Plant commissioning then proceeded but the progression to active commissioning, planned for December 1992, was delayed by a protracted public consultation exercise to satisfy recent environmental legislation. Government and Regulatory approval to proceed was finally given, leading to the commencement of active commissioning on 17 January 1994.

Design Intent

All BNFL's operations on the Sellafield site are strictly controlled by Government Authorising Bodies. For the control of discharges to the environment, the relevant bodies are Her Majesty's Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF). Certificates of Authorisation issued by the Departments impose strict conditions on the discharge and disposal of radioactive waste. They also require BNFL to carry out detailed environmental radioactivity monitoring programmes (in addition to their own independent measurements) and to use "Best Practicable Means" to further limit the radioactivity in the waste discharged.

BNFL's strategy for the management of aerial and marine radioactive effluents arising from THORP operations were first formally stated at the Windscale Inquiry in 1977. In summary the design intent was that:

(a) radiation exposure of members of the general public, arising from discharges from THORP and associated facilities, should be "As Low As is Reasonably Practicable" (ALARP);
(b) in any case, no member of the most highly exposed (critical) group should receive an annual committed effective dose equivalent (CEDE) in excess of 50μSv from either aerial or marine discharge routes.

In the context of exposure to the general public, it is important to consider both:

(a) individual risk, as measured by doses to members of the critical group, and
(b) societal risk, as measured by the collective dose to large populations over a long time period.

In addition to dose uptake constraints, HMIP has more recently set an Aerial Discharge Authorisation for THORP which limits the quantity of specific radionuclides which can be released from the THORP stack on a daily, quarterly or annual basis. The annual discharge limits for the most significant radionuclides are given in Table 1.

As far as liquid discharges are concerned, HMIP's Authorisation relates to the whole site rather than THORP specifically and is depicted in Table 4.

Scope of Paper

This paper describes how the design intent on aerial and liquid discharges has been met. Concentrating primarily on aerial discharges, it examines the different sources of off-gas arisings from the THORP process and describes how the different types of stream have been segregated for treatment prior to extraction and discharge from a common 125m stack. The development work associated with some of the key process equipment is described and key design features of the equipment are covered.

As far as liquid discharges are concerned, the paper describes the overall philosophy behind the THORP flowsheets which allows the environmental impact target to be met. The treatment of some waste streams is undertaken in dedicated equipment within the THORP building. Other streams are mixed with effluents from elsewhere on site and treated in a variety of existing or new site facilities, which are described in outline.

Finally, the paper compares the predicted aerial and liquid discharges from the plant with the latest Discharge Authorisations to illustrate how well the plant design is expected to meet its objectives.

Table 1 - Comparison of Predicted THORP Aerial Discharges with Discharge Authorisation

Radionuclide	THORP Discharges (TBq/yr)	THORP Downstream Plant Discharges (TBq/yr)	THORP[1] Total Discharges (TBq/yr)	THORP Authorisation (TBq/yr)
H-3	21.6	0.0007	21.6	43
C-14	0.434	~0	0.434	0.87
Kr-85	369,000	6.61	369,007	470,000
Sr-90	0.004	0.0036	0.0076	0.0078
Ru-106	0.0355	0.0016	0.0371	0.05
I-129	0.0218	0.0031	0.0249	0.044
Cs-137	0.0055	0.005	0.0105	0.011
Pu(Alpha)	0.00027	~0	0.00027	0.0005
Total Alpha	0.00048	~0	0.00048	0.001
Total Beta (β5)	0.152	0.017	0.169	0.28

Critical Group dose [1] = 22μSv/yr
Target dose = 50μSv/yr

(1) Based on 1200t(U)/yr of reference fuel

GASEOUS ARISINGS

THORP is an integrated reprocessing plant with all the major process steps being carried out in a single building (See Fig 1). Only where use could be made of waste and effluent treatment facilities serving existing process plants on the Sellafield site are certain operations performed outside the THORP building. The THORP process breaks down principally into a shear/leach "Head End" section with clarification of the dissolver solution, followed by chemical separation of the uranium, plutonium and fission products by solvent extraction in pulsed column contactors. The uranium and plutonium products are purified by further solvent extraction prior to conversion to solid forms suitable for storage or return to customers.

Within the plant as a whole, a variety of different off-gases arise.

Dissolver Off-gas

Leaching of the sheared fuel in boiling nitric acid releases virtually all the volatile components of the irradiated fuel assemblies, principally I-129, C-14 and Kr-85. Also Nitrogen oxides (NOx) are generated by dissolution of the fuel. The dissolver off-gas (DOG) stream consists of the ventilation air from the shear machine and the off-gases from each of three dissolvers, amounting to a flow of about $400m^3$/hour. The stream also contains very fine particles of fuel dust which are not captured in the dissolver and, in total, represents by far the biggest off-gas challenge in THORP.

Central Off-gas

Apart from the dissolvers, most of the other vessels and process equipment within the Head End and Chemical Separation Plant are vented via a separate Central Off-gas (COG) system. This has to accept a large number of streams with widely differing flows and compositions.

The most environmentally significant component of the COG arisings is I-129. Although the vast majority is volatilised into the DOG system, any residual amount is assumed to be released from downstream vessels into the COG system. In addition, aerosols created in the chemical treatment and liquor transfer processes throughout the plant carry particles of non-volatile actinides and fission products into the ventilation stream. The nuclides that contribute significantly to the aerial activity levels are Ru-106, Pu, Sr-90 and Cs-137.

There are certain non-active arisings which have to be considered in the design of the COG system. The only one subject to regulatory discharge limits is NOx.

Cell Ventilation

THORP is designed on the principle of cascading depressions between areas to provide barriers against spread of contamination. Cells and caves which contain the most highly active processes in the plant are therefore maintained under a depression with reference to the adjoining access areas. Generally inleakage at cell depression is adequate to provide the air flow required in a cell with little or no heat emitting equipment. In cells with large heat sources, additional air is provided to dissipate the heat by purpose-built engineered inlets comprising HEPA filters and control/fire dampers.

Air extracted from cells and caves is assumed to exhibit the same distribution of radionuclides (footprint) as is present in the process equipment in the cell. Arisings from mechanical operations of fuel shearing in head end caves have been assessed on empirical data from similar operations on other plants. In cells where operations are normally clean (washdown provisions are installed to clean-up spillages), the arisings have been based on potential re-suspension of liquor from pools on the bunded floors.

Building Ventilation

Building ventilation and heat systems provide a comfortable working environment in 'normal' and 'limited' access areas of the plant.

Normally accessible (unrestricted access) areas, despite having no definitive arisings (except under abnormal circumstances), are allocated an assumed mean airborne contamination which, if inhaled over a 2000 hour year, would result in 1% of the whole body dose target being received by inhalation.

Restricted access areas normally have very low arisings but have the potential for activity being present for short periods in abnormal circumstances. Such areas are allocated an assumed mean airborne contamination of ten times that allocated to the normal access areas. Under such conditions personnel would wear respiratory protection. In each instance the nuclides present are in proportion to the 'footprint' of the active material being processed.

Total Ventilation System

A schematic diagram showing the inter-relationship of the various components of the THORP ventilation system is given in Figure 2. It illustrates that main ventilation streams are kept separate until they enter the 125m stack from which they are discharged.

FLOWSHEET AND EQUIPMENT DEVELOPMENT

Dissolver Off-gas

The prime task of the DOG system is to remove nitrogen oxides generated by the dissolution of the UO2 fuel, together with the major volatile radioactive species released as the fuel is dissolved. The DOG challenge is illustrated in Table 2, together with the flowsheeted Decontamination Factors (DFs) for each item of equipment.

Table 2 - DOG System Performance

Radionuclide or species	Arising(1) (TBq/yr)	Flowsheet Decontamination Factor (DF)				
		Condenser	Acid Scrubber	Caustic Scrubber	Weak Acid Scrubber	HEPA
H-3	97.2	1	3	1.5	1	1
C-14	28.9	1	1	70	1	1
Kr-85	3.69E5	1	1	1	1	1
Ru-106(gas)	37.5*		20	100	1	1
Ru-106 (solid)	37.5*		1	1	1	10E4
I-129	1.41	1	1.05	100	1	1
NOx	8.1E4 m^3	1.5	3	100	1	1
Fuel Dust	2.6E3 kg	25	70	1.2	1.4	10E4

* post condenser (1) Based on 1200t(U)/yr of reference fuel

Recognising the need for highly efficient I-129 removal in THORP, the first requirement is to volatilise as much as possible at the dissolver stage. Once converted to organic forms by contact with solvent (TBP/OK) further down the process, iodine capture is much more difficult. Experimental work has demonstrated that the NOx evolved during dissolution of the fuel is effective in maintaining the I-129 in its volatile molecular form, which renders it easier to remove from the off-gas system.

Treatment of the DOG stream is performed principally by acid and caustic scrubbing in packed columns. The full train of DOG equipment is shown in Fig 3

The off-gases from each dissolver pass through their own dedicated reflux condenser and are then combined to be fed to the recombined acid column. Here, scrubbing with 6M nitric acid serves to remove the bulk of the oxides of nitrogen and two impingement plates at the base of the column assist removal of the entrained fuel dust.

A small amount of the iodine is also absorbed in the acidic scrubber liquor and this has to be removed before the liquor is fed back into the main process stream. This iodine desorption is achieved by heating a bleed of scrubber liquor and passing it through a small diameter packed column, counter current to a flow of air, which then joins the main off-gas flow carrying the desorbed iodine.

Following the recombined acid column is a large empty vessel, termed the " noxidiser", which provides a residence time of about two minutes in which the nitric oxide present is substantially oxidised to nitrogen dioxide by the oxygen in the air (which makes up the major proportion of the off-gas flow).

Increasing the proportion of nitrogen dioxide in this way improves the absorption of the nitrogen oxides in the caustic scrubber immediately downstream. The caustic scrubber will achieve a reduction in the nitrogen oxide concentration to below 1000ppm and a reduction in the I-129 and C-14 by factors of 100 and 70 respectively. Following the caustic scrubber the gas stream will be scrubbed with chilled recirculating water to remove entrained caustic soda droplets and dehumidify the gas. Finally the DOG stream undergoes two stages of HEPA filtration for final particulate removal before discharge to atmosphere.

DOG Supporting Development

In view of the dependence of the off-gas system upon aqueous scrubbing, the major facility in the development programme was a pilot scale rig - the dissolver off-gas pilot plant - to enable study of each of the scrubbing processes in the DOG system.

The rig, depicted in Fig 4, consisted of a small packed column, 100 mm diameter, with a packed height of 3m. Equipment was provided to generate a suitable mixture of oxides of nitrogen in air to which gaseous iodine could be added as appropriate. The scrub liquor could either be nitric acid or caustic soda solution and the liquor recirculated continuously through the column or treated in a single pass.

Sampling of the gases by means of heated sample lines enabled gas phase analysis by UV or IR spectrometers for nitrogen oxides and carbon dioxide respectively. Sampling for iodine was achieved by selective absorption on heated beds of silver nitrate impregnated absorbers.

As originally conceived the pilot plant was to have handled iodine traced with I-131 in order to facilitate analysis to the low levels required. This increased significantly the complexity of the rig and would have made experimental work much more unwieldly. However, satisfactory development of ion selective electrode techniques sensitive to iodine concentrations as low as 1×10^{-7} M removed the need for trace active work and considerably simplified the experimental programme. The outstanding analytical difficulty was the determination of inactive iodine on the silver nitrate absorbers used for sampling the gas streams but this was readily overcome by the use of neutron activation analysis.

Experimental investigation of acid and alkaline scrubbing of nitrogen oxides was supported by work on computer simulation of the processes in a package developed by BNFL. This package, NOXSIM, was thoroughly validated against the pilot plant experimental results and thus could be used with confidence in the design of the THORP off-gas columns.

Apart from the development work on the DOG pilot plant, other supplementary development was undertaken to address specific issues within DOG system. For example, experiments were performed to investigate iodine behaviour during dissolution of irradiated fuel and the subsequent treatment of dissolver off-gases. These investigations were part of the experimental work carried out on the THORP miniature pilot plant, a highly active 1:6500 scale pilot plant built primarily to study solvent extraction flowsheets. The dissolution of irradiated fuel served to show that the I-129 was present as molecular iodine and hence the simulation in the DOG pilot plant was valid.

Carbon-14 behaviour was less of a problem than I-129 since the bulk of it would be present as carbon dioxide and this would mix with the much larger quantities of CO_2 in air. There is abundant information on caustic scrubbing of CO_2 in air but a number of confirmatory experiments were necessary on the DOG pilot plant to ensure that the presence of 1 to 2 per cent NOx would not affect the CO_2 removal.

Not all the C-14 released from the dissolver is in the form of CO_2: a small amount is present as CO. Measurements performed during dissolution of irradiated fuel on the THORP miniature pilot plant confirmed literature information that the CO component was less than 1% and would not compromise the ability of the caustic scrubber to reduce C-14 levels by a factor of 70.

Measurement of CO_2 absorption has also proved important in the detection of potential maloperation. Experiments with the DOG pilot plant have shown that monitoring for increased CO_2 in the off-gases downstream of the scrubber provides adequate warning of approaching loss of alkalinity.

One issue raised well into the design phase of the THORP DOG system was a suggestion that troublesome quantities of volatile Ru-106 might penetrate into and even through the off-gas system. At first this question was explored by means of dissolution of irradiated fuel in a highly active cell and then in greater depth using trace active laboratory experiments. These two pieces of work were sufficient to indicate that only relatively small amounts of volatile Ru-106 would be generated and that the off-gas system as designed would ensure adequate removal. Subsequent studies with irradiated fuel dissolution in the THORP miniature pilot plant have indicated that the removal efficiency in the THORP dissolver off-gas system will be ten times greater than initially estimated and thus the impact of volatile Ru-106 has proved to be of little significance.

Central Off-Gas

As already described, the COG system serves most of the vessels and equipment downstream of the dissolvers. The full train of COG equipment is shown in Fig 5. The off-gas arisings from different parts of the plant or from different types of equipment are combined into a series of "headers", which feed into the COG system at an appropriate point according to the type of decontamination required. The arising of key components in the COG feed is shown in Table 3, together with the flowsheeted DFs for each item of equipment.

Table 3 - COG System Performance

Radionuclide or species	Arising[(1)] (TBq/yr)	Flowsheet Decontamination Factor (DF)			
		Caustic Scrubber	ESP	Dehumidifier	HEPA
Sr-90	56.4	10	100	10	10E4
Ru-106(gas)	0.15	1	1	5	1
Ru-106(solid)	14.4	10	100	10	10E4
I-129	0.009	2	1	1	1
Cs-137	79.9	10	100	10	10E4
Pu(alpha)	6.19	10	100	10	10E4
NOx	84.9 m^3	43	1	1.6	1

[(1)] Based on 1200 t (U)/yr of reference fuel

First in the COG system is a 1.8m diameter caustic scrubber which serves the iodine header. This is primarily for iodine removal though it removes NOx and some particulate as well. A conservative Decontamination Factor (DF) of 2 has been assessed for iodine removal because the residual iodine from these downstream operations will be substantially organic. The scrubber performs a secondary function as an iodine trap in the event of accidental neutralisation of iodine-bearing streams in the Low Active Effluent Treatment Plant.

The next equipment in the train is a pair of 680 m^3/hr Electrostatic Precipitators (ESPs) whose purpose is the removal of entrained droplets and aerosols from the particulate header. These devices have proved to be very reliable and effective in other applications in highly active plant at Sellafield and only generate a low volume of liquid waste for treatment. Furthermore, by reducing the load on the HEPA filters at the end of the train, they reduce filter change frequencies and waste disposal costs. However care has to be taken to ensure that any solvent vapour in the feed stream is well below its lower flammability limit because the high voltage across the electrodes represents a potential ignition source.

The dehumidifier scrubber downstream of the ESPs is primarily aimed at removing moisture from the gas stream prior to HEPA filtration. This would cause deterioration of the filter elements, leading to their eventual collapse and hence increased solid waste arisings. In order to prevent this the vent gas has to be cooled below its dew point, using a chiller unit to cool the dehumidifier scrubber liquor.

In addition, the dehumidifier acts as the primary disentrainment device for streams for which treatment in the ESPs is unnecessary. These include the low contamination vent header and the pulse header from the pulsed column drive legs in the highly active and plutonium purification cycles. With a large number of pulse systems operating independently the header could, theoretically, receive a very high flowrate if all the pulse systems discharged simultaneously. A Monte Carlo approach was used to predict the highest realistic flow likely to arise, thereby minimising the degree of over- design.

The final equipment in the train are banks of primary and secondary remote-change HEPA filters, which act as a final polishing step for particulates and as a back-up in the event of equipment maloperation upstream. Hot air is bled into the vent before the filters to ensure that the feed is well above its dew point.

The COG stream is finally discharged from a 125m stack, together with other THORP off-gases.

COG Supporting Development

In comparison with the DOG system, very little specific development work was necessary to support the COG system design. In relation to scrubber design, data generated from the DOG pilot plant were used, with the help of the NOXSIM computer model, to assist design of the COG unit.

One area which did require special attention was the characterisation of the aerosols generated by pulsing systems, agitation systems and various liquor transfer devices within the plant. Information on quantity and droplet size distribution was generated by development trials on replica equipment. This work showed that treatment of many of these streams in the ESPs was unnecessary; scrubbing in the dehumidifier gave sufficient decontamination.

Some development was performed on the ESP units during the design phase. Experience on earlier plants and collaboration with the manufacturer enabled a simpler and more compact design to be used on THORP without losing any efficiency. The simplification of the central electrode design within the collection tubes was demonstrated on a test rig.

Glovebox Ventilation

The finishing processes on the purified plutonium stream are largely conducted in glovebox containments. As gloveboxes with their high alpha inventories have to be protected against both overpressure and excessive depression under fault conditions, they are separately vented from the COG system.

The glovebox extracts are fitted with vortex amplifier (VXA) control devices operating under ambient pressure control to provide a controlled normal operating depression with small gas purge flows. VXAs and other fluidic devices are used extensively in THORP because their maintenance-free operation is invaluable in a highly radioactive environment. If a glovebox containment develops a leak or is breached the VXA device automatically opens to provide enhanced containment flow through the breach. These devices have proved highly reliable on both plutonium glovebox and alpha laboratory analytical suites in existing Sellafield facilities.

The glovebox extracts are collected together and pulled through primary and secondary HEPA filters prior to discharge up the 125m stack in a dedicated flue.

Cell and Cave Ventilation

The plant is designed on the principle of segregation of ventilation streams of different activity levels to restrict the spread of contamination and minimise the amount of highly active waste materials.

Cell and cave air is extracted through fully shielded ducting to two active filter caves located in the centre building where the streams are filtered through 2-stages of HEPA filtration prior to discharge through the main stack. Several cells with minimal contamination potential have local manual-bag-change, circular filters, as do cells with predominantly alpha-bearing streams.

To prove the concept of the filter cave, prototype filter modules were constructed to minimise the pressure drop and improve the isolation damper maintenance problems. The grab assembly, which is essential for reliable filter change operations, underwent several revisions which were extensively tested to ensure the action was positive and reliable over a long period in a hazardous environment.

Building Ventilation

The largest proportion of THORP building is devoted to normally accessible work-areas and interconnecting corridors. The air required for ventilation purposes is supplied filtered at a constant temperature of 19°C in winter. Overall air change rates are inappropriate for use on large plants and would have resulted in an expensive and uneconomic installation. Each ventilated space within the THORP building has had its requirement for fresh air assessed on the need for heat, fume or humidity removal, occupation level and function. This approach has resulted in a smaller, more economic ventilation plant able to meet the large variations in individual room requirements.

To maintain the principles of containment, air is transferred from zones of low contamination potential to zones of higher contamination potential, thus minimising the spread of activity. Air extracted from the various zones is collected in separate dedicated ducting systems. In the case of the building air from beta-gamma areas, this is discharged unfiltered from short ducts terminating above the building parapet height. Air from alpha areas where there is normal access is passed through single stage HEPA filters before discharge from the main stack.

Air from limited access areas is extracted through single-stage HEPA filters in manually changed (bagged out) housings and discharged to atmosphere from the main stack. Limited access alpha areas are filtered through two stages of HEPA filters before stack discharge.

Stack Dispersion Modelling

The first detailed flowsheets identified a requirement for a stack capable of discharging 485,000 m^3/hour of effluent at a discharge height of between 80m and 120m based on a building height of 35m. A multi-flue connected to the main plant (centre building) was proposed, the main ducts being contained within a bridge section which would be used for flow measurement and sampling/monitoring.

Wind tunnel tests were carried out to assess the performance of both stacks (80m and 120m) on a 1:200 scale site model, with varying windspeeds and directions and a range of discharge flue velocities. Due to the lack of sufficiently accurate gas concentration measurement equipment at this time, much of the work was done using high speed camera techniques with a neutral buoyancy smoke tracer in the flue discharge. The 120m high stack, employing a discharge velocity in excess of 30m/sec gave an effective height of 120m based on the visual appearance of the plume.

Further wind tunnel tests were carried out in 1988 to verify the performance of the stack based on final flowsheet values and using modern Flame Ionisation Detector systems which can detect levels an order of magnitude smaller than the earlier technology. In the interim period, the stack design had been simplified to one central flue of 3m diameter housing the major ventilation streams but incorporating COG and DOG streams at 108m level. The glovebox discharge joined the main flue at 108m level but continued inside the flue in its own discharge pipe to prevent back-pressure.

The later tests were used as a basis for quantitative effective height assessment, and demonstrated an effective height of only 85m for a 120m stack when buildings are present and in the most limiting weather conditions. Further tests with the model were able to show that an additional flue extension of 5m to give a height of 125m would produce an effective discharge height of 92m, which was sufficient to keep the critical group dose impact well below the target value and ensure the plume clears all site buildings. Thus the final design has a flue discharge height of 125m, a windshield roof level of 115m and a flue velocity of 34m/s.

LIQUID ARISINGS

Flowsheet Philosophy

Being the first new reprocessing plant at Sellafield for nearly 30 years, THORP was able to take advantage of modern thinking in flowsheet design, particularly with respect to minimisation of waste arisings. A number of clean technology concepts have been incorporated into the overall process in order to simplify waste treatment and reduce arisings.

A major improvement in the THORP flowsheet compared with earlier Magnox flowsheets is the move to salt-free reagents in the plutonium reduction stage. Traditionally, a ferrous salt such as ferrous sulphamate has been used to reduce plutonium to the non-extractable 3-valent state in order to effect a separation from uranium. This imposes severe restrictions on subsequent waste stream concentration because of the presence of iron salts. Process chemistry development for THORP identified U IV stabilised by hydrazine as the optimum reductant. Whilst not strictly salt-free, the added uranium is oxidised to U VI and follows the product stream with no impact on waste stream composition.

Similarly, the oxidising and reducing agents used in the downstream purification cycles are salt-free reagents, giving rise only to water and/or simple gases which have no deleterious effects on subsequent effluent treatment processes. Thus an even greater proportion of active waste isotopes can now be concentrated and subsequently vitrified. Apart from the raffinates from solvent washing, for which no satisfactory substitutes for sodium carbonate and sodium hydroxide have yet been found, all the aqueous process effluents containing significant levels of radioactivity can be so treated.

The clean technology concept of effluent recycle has been used extensively in THORP. This is particularly true of the use of recycled nitric acid for many process feed streams. Acid recovery takes place in the salt-free evaporator system which processes all salt-free waste streams from the THORP process apart from the highly active raffinate from the first cycle (see below). The salt-free evaporator contains a fractionating column and produces three products, a concentrate containing virtually all of the activity, low acidity overheads which are discharged to sea and a low active recovered acid stream (6M HNO_3) which is recycled to the main plant. This recovered acid is used in virtually all acid feeds to THORP apart from those to the dissolver.

Breakdown of Liquid Effluents

There is a large number of liquid effluents arising at different points within the THORP process. The purpose of this paper is not to describe them all in detail but to give an overview of the types of streams and the methods of treatment adopted. THORP effluents can be categorised into three general types:

a) those which are normally suitable for sea discharge without activity reduction
b) those which are treated locally within THORP
c) those which are treated centrally by site facilities

Discharges to the Segregated Effluent Treatment Plant (SETP)

Over 30 streams from THORP are directed to SETP for sea discharge. Acidic and alkaline streams are segregated within THORP and transferred separately to SETP, where the acidic effluent is neutralised before mixing with alkaline effluent. Care has to be taken to avoid any risk of acidifying alkaline streams which contain Iodine-129, as this would probably cause volatilisation and release of this environmentally significant radionuclide. The combined effluents are then sentenced and discharged to sea.

Typical feeds to SETP include decontamination liquors from the THORP decontamination centre, condensates from the MA salt-free evaporator, excess recovered nitric acid, condensate from UN evaporation, COG caustic scrubber liquor and many others.

Effluents treated locally in THORP

Certain effluents require a dedicated treatment facility local to the source of arising in order to reduce the concentration of specific radionuclides before sea discharge. The first of these is the DOG caustic scrubber effluent, which requires removal of carbon-14 and plutonium present as fuel dust. As the carbon-14 is present as carbonate, along with absorbed atmospheric CO_2, the selected removal process uses precipitation as barium carbonate. This process was developed in the laboratory to establish optimum settling time, and washing parameters for the precipitate, together with an understanding of the behaviour of small particles of fuel dust in the process. The precipitate is encapsulated in cement as an Intermediate Level Waste and the low active supernate is sentenced and discharged to sea.

The second stream requiring local treatment is the Feed Pond purge. Feed Pond operations may result in suspension of some solids bearing cobalt-60 which would have a significant environmental impact. Therefore the purge is passed through a Funda filter for solids removal before discharge to sea. It is noteworthy that pond water treatment for THORP is much simpler than for the earlier Magnox plants. Corrosion of stainless steel or Zircaloy cladding on oxide fuel is much less than for Magnox alloy, and the spread of any corrosion products is limited by intermediate containers. Hence ion exchange treatment of THORP pondwater is unnecessary.

The final category of effluents requiring local treatment in THORP are the Medium Active salt-free effluents. These are typically the raffinates from the product purification stages of the chemical separation plant, which are relatively high in activity but low in salts thanks to careful flowsheet design. These streams can be evaporated by a substantial factor (~50) without risk of crystallisation or precipitation in the concentrate. The overheads from the Medium Active salt-free evaporator are fed to a fractionating column for acid recovery, whilst the concentrate is fed to the site HA evaporator. This is a good example of clean technology by recycling streams within the overall process.

Effluents Treated by Site Facilities

Where suitable site facilities already exist, these are utilised for the treatment of appropriate THORP effluents. The facilities in question are:

(i)	HA liquor evaporation and storage	(HALES)
(ii)	Evaporation of salt-bearing liquors	(Salt Evaporator)
(iii)	Enhanced Actinide Removal Plant	(EARP)
(iii)	Solvent Treatment Plant	(STP)

The HA evaporator takes the first cycle aqueous raffinates from reprocessing, which bear over 99% of the fission products and minor actinides, and concentrates them by a factor of 20-100 for interim storage in high integrity, double-walled stainless steel tanks. The stored concentrate is subsequently converted by vitrification into monolithic glass blocks encased in stainless steel cylinders, which are suitable for long term storage and ultimate disposal.

The salt evaporator deals principally with solvent wash raffinates. The organic solvent (TBP/OK) used in the main chemical separation process becomes contaminated and degraded over time, so is chemically washed and recycled into the process in order to minimise waste arisings. The solvent wash raffinates carry a heavy burden of sodium hydroxide and carbonate from the washing process, so are unsuitable for evaporation with other salt-free MA effluents. Hence they are neutralised and concentrated in a separate salt evaporator, the concentrate being delay-stored for ruthenium decay prior to treatment in EARP.

There are some aqueous effluents which are too voluminous for evaporation but contain concentrations of certain radionuclides, especially actinides, which make them unsuitable for direct sea discharge. Most of these streams arise in the Magnox plant and are acidic iron-bearing streams. This feature has been used in the development of the Enhanced Actinide Removal Plant (EARP - see Fig 6). The streams are neutralised with caustic soda to produce a ferric hydroxide floc which carries with it virtually all the actinides. Addition of small quantities of specific chemicals also improves beta decontamination. The floc as first produced is too voluminous and needs to be de-watered for economic storage and disposal. This is achieved by ultrafiltration, giving typical concentration factors of ~500. The concentrated floc is encapsulated in cement ready for disposal and the permeate from ultrafiltration, being very low in radioactivity, is discharged to sea. Only a small number of streams from THORP are fed to EARP; these include certain solvent washes from the plutonium purification cycle, Multi-Element Bottle (MEB) flushings from the Feed Pond and salt evaporator concentrate (which contains arisings from Magnox and THORP).

Waste solvent arisings are minimised by the use of solvent washing and recycle. However the two sources of arising in THORP are the HA cycle solvent purge and float-off solvents arising from end-of-campaign washouts. Waste solvents from Magnox and THORP reprocessing are currently stored pending completion and commissioning of the solvent treatment plant (STP - See Fig 7). The process at the heart of STP is alkaline hydrolysis. Boiling the spent solvent with concentrated caustic soda breaks it down into three phases: an organic kerosene phase which can be burned; a sodium dibutyl phosphate phase (Na DBP) which is low enough in contaminants for sea discharge; and an aqueous sodium hydroxide phase containing the vast majority of the radioactivity which can be combined with aqueous wash streams from the pretreatment stages and fed to EARP for clean up and discharge.

PREDICTED DISCHARGES

Gaseous

A comparison of the flowsheeted discharges with the latest Discharge Authorisation is given in Table 1, based on the design throughput of 1200t(U)yr of reference fuel. For all radionuclides with a defined limit, there is a reasonable margin between the flowsheeted discharge and the authorised limit. In terms of the Critical Group impact, the predicted dose under flowsheet conditions is 22μSv/yr, against the target dose of 50μSv/yr.

Liquid

As mentioned previously, THORP does not have its own Liquid Discharge Authorisation so Table 4 compares the predicted THORP discharges, both direct and via downstream treatment plants, with the total site Discharge Authorisation. In general, it can be seen that the predicted discharges are only a small percentage (generally <10%) of the total site allocation. For the Critical Group impact, the predicted dose is 36μSv for local seafood and shellfish eaters. This group is distinct from the aerial discharge Critical Group, so can be compared in isolation with the target of 50μSv.

Table 4 - Comparison of THORP Liquid discharges with Total Site Authorisation (TBq/yr)

Radionuclide	Total Predicted THORP Discharges	Site Authorisation
H-3	6,972	31,000
C-14	0.497	20.8
Co-60	2.37	13
Sr-90	0.844	48
Ru-106	18.2	63
I-129	1.40	2
Cs-137	6.07	75
Pu (alpha)	0.045	0.7
Total alpha	0.096	1
Beta-5	30.6	400

Critical Group dose = 36 μSv/yr   Target dose = 50 μSv/yr

CONCLUSIONS

1 BNFL's Thermal Oxide Reprocessing Plant incorporates a variety of modern techniques for the treatment of gaseous and liquid effluents.

2 For off-gases, the design target has been met by subdivision of the arisings into discrete treatment trains, viz: dissolver off-gas, central off-gas, gloveboxes, cells and caves, and building ventilation.

3 Each stream is treated by a combination of process steps appropriate to its composition and the required degree of decontamination.

4 Development work on key parts of the off-gas treatment system, supported by theoretical modelling, has ensured that plant design has a firm foundation.

5 All active streams are discharged, after treatment, from a 125m stack whose effective height has been demonstrated to be 92m.

6 Modelling work predicts that the critical group dose from THORP aerial discharges will be 22μSv compared with a design target of 50μSv.

7 The treatment of liquid effluents has been simplified by the use of salt-free reagents and recycling of waste streams within the process.

8 Liquid effluents are either treated by dedicated equipment in THORP or combined with other streams for treatment in common site facilities, prior to sea discharge.

9 The treatment process used depends on the volume, nature and specific radionuclide content of the effluent.

10 The net result is a predicted dose to the marine critical group of 36μSv, against the design target of 50μSv.

11 The individual radionuclide discharges from THORP generally will make only a small contribution to the total site liquid discharges.

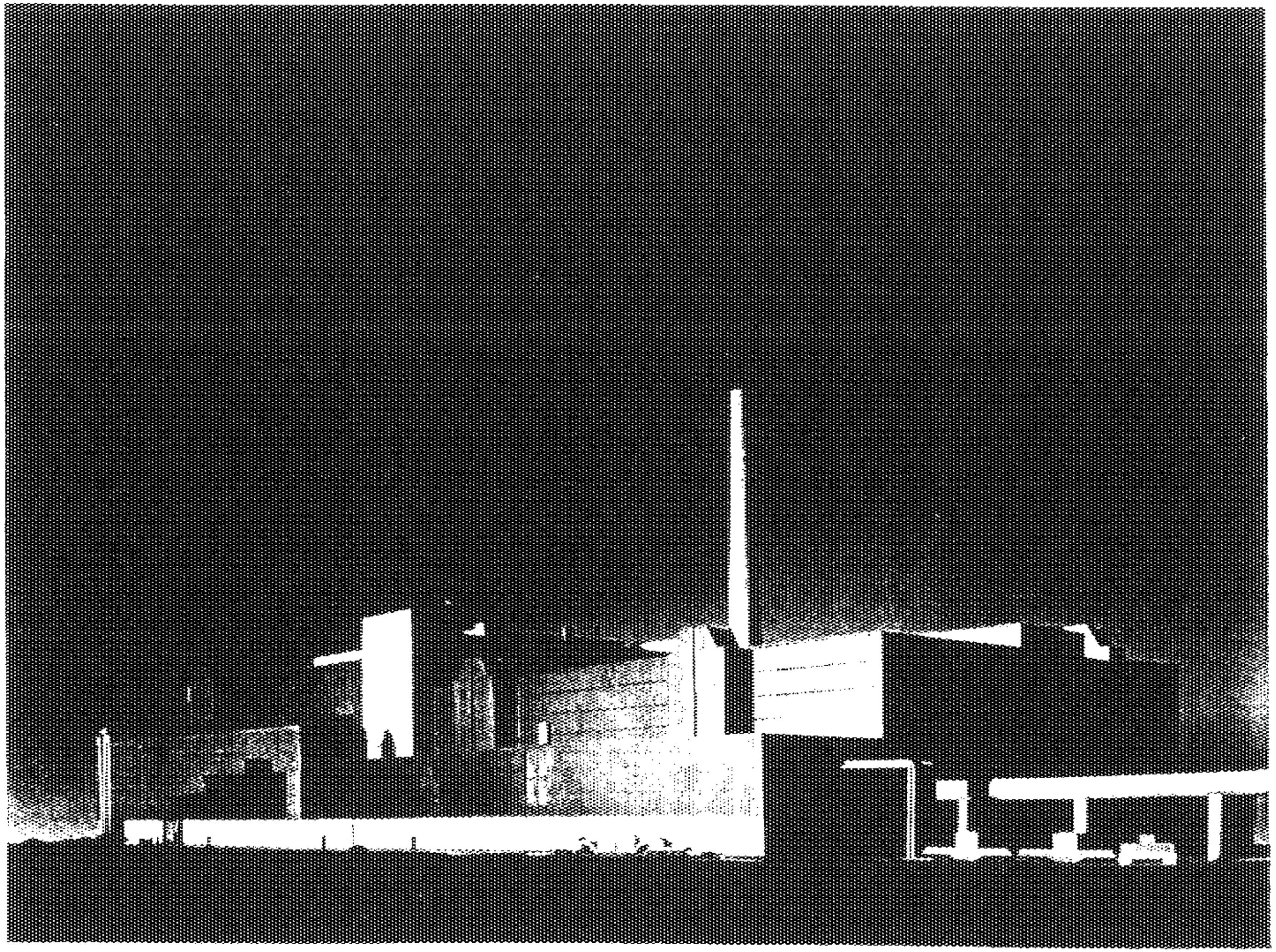
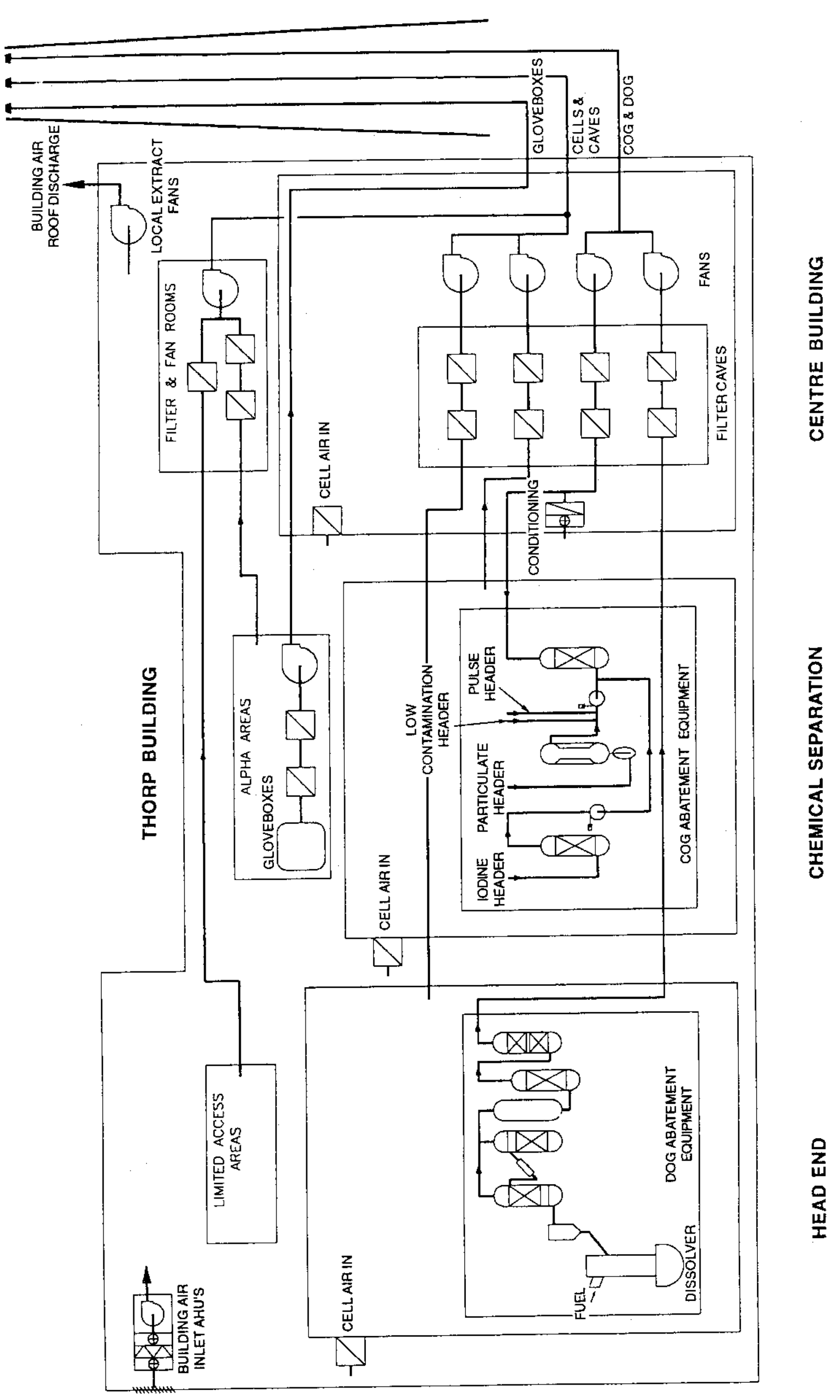


Figure 1 THORP building



BNFL

Figure 2 Schematic of THORP active ventilation systems

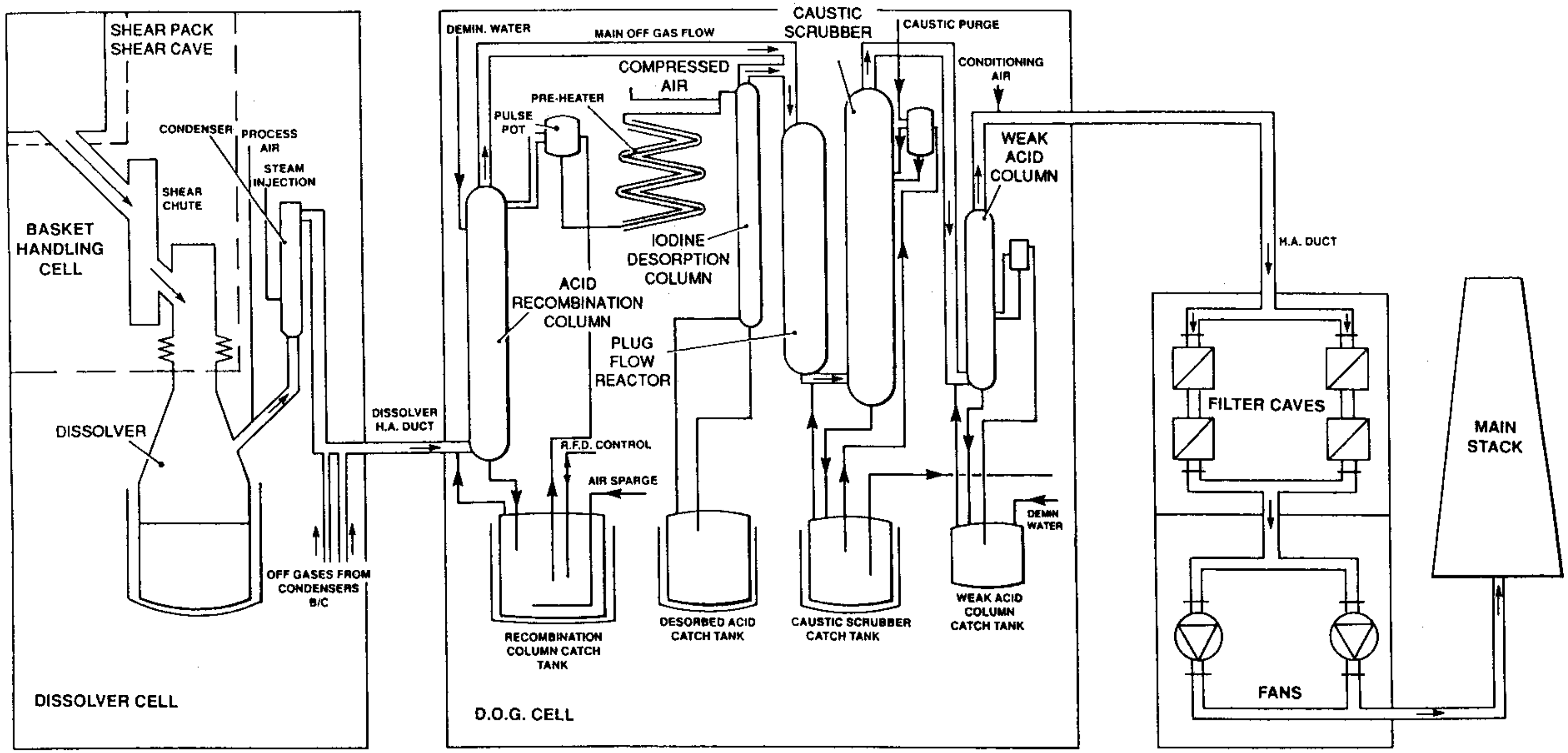


Figure 3 Dissolver off gas (DOG) extract system

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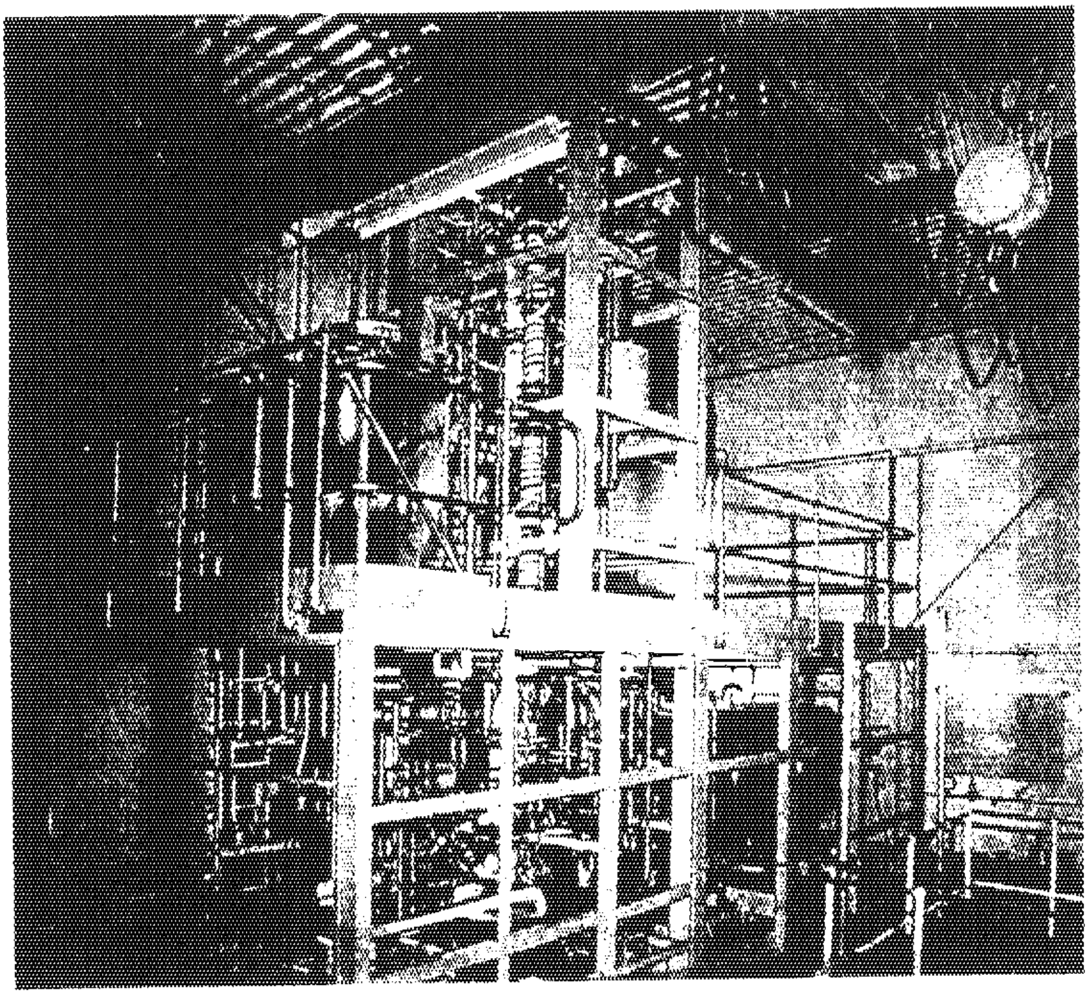


Figure 4 Dissolver off gas pilot plant

BNFL

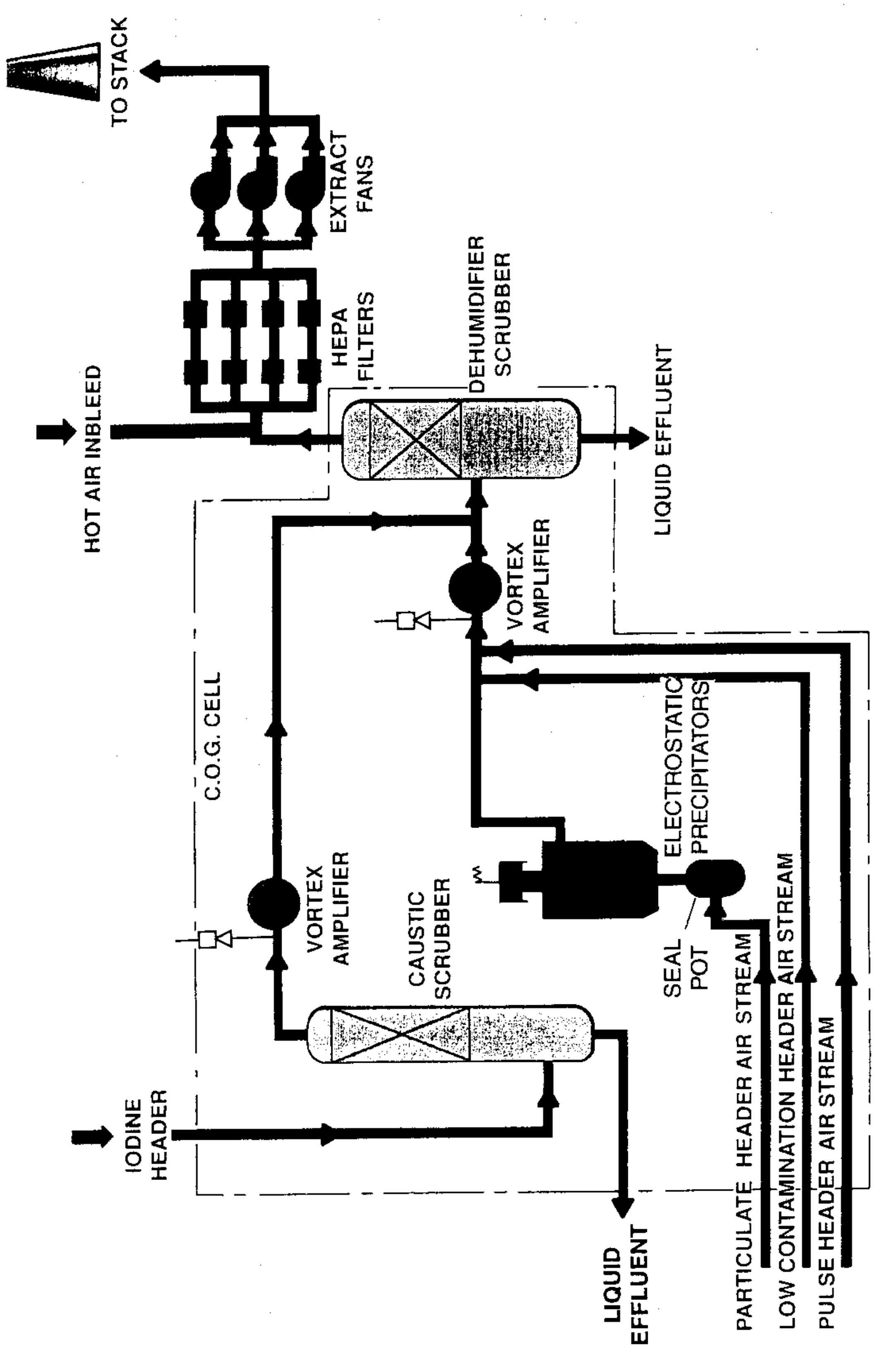
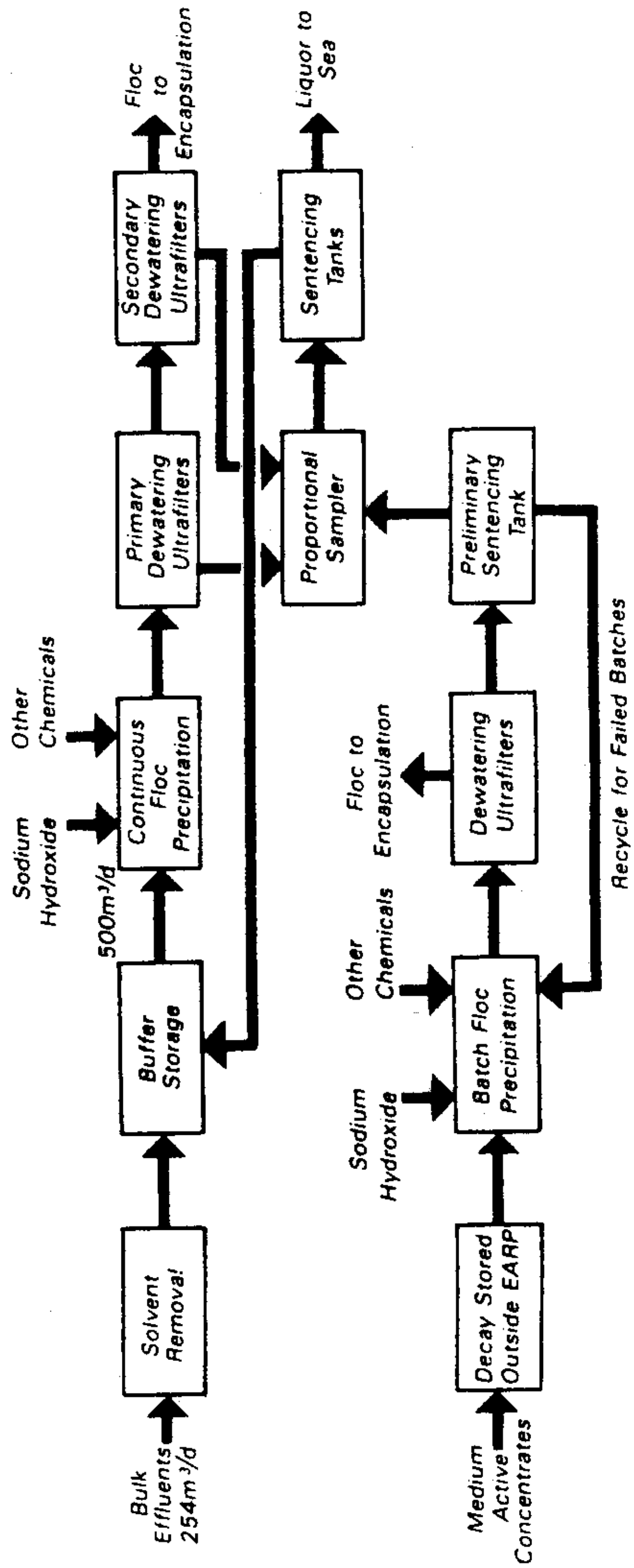


Figure 5 Central off gas (COG) extract system

BNFL



BNFL

Figure 6 EARP - Simplified Process Flow Diagram

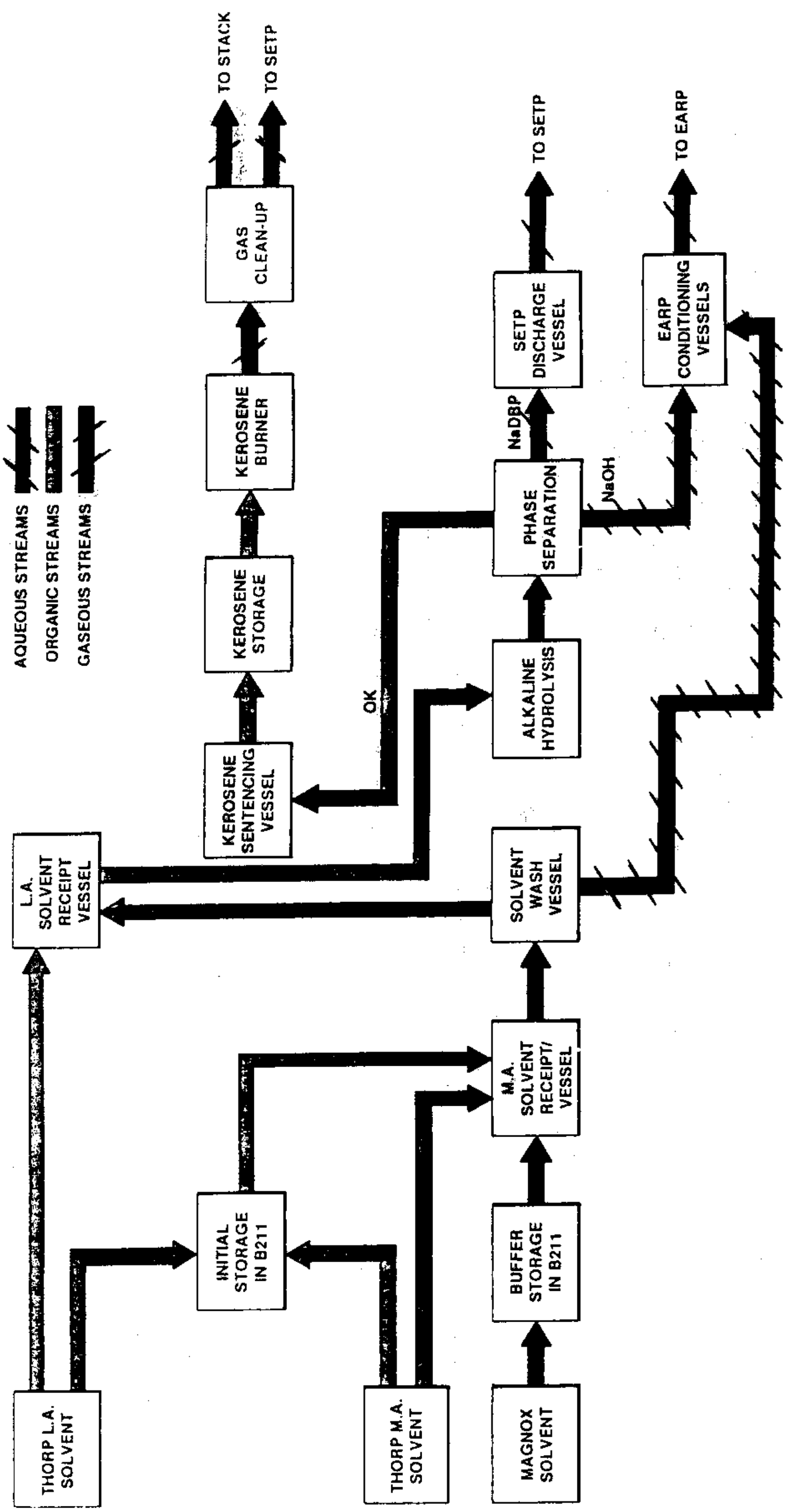


Figure 7 STP Block Flow Diagram

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THE MINIMIZATION OF RADIOACTIVE RELEASES FROM THE LA HAGUE PLANTS

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1. INTRODUCTION

The Environmental Rio Summit held in 1992 revealed the commitment of industrial countries in favor of sustainable development. France's long-standing reliance on nuclear energy has been for years a confirmation of such a commitment, which led to drastic reductions of toxic emissions and to an improvement of air and water quality even in advance on the Rio decisions. Among the attractive features of nuclear energy is the absence of carbon dioxide and other greenhouse gases release into the atmosphere. The early choice of a closed fuel cycle is fully consistent with such a policy. Spent fuel reprocessing significantly contributes to environment protection and conservation of natural resources : waste are safely conditioned in a form adapted to their nature and recycling of energetic materials allows to recover the equivalent of nearly 20,000 TOE per metric ton of spent fuel.

Also consistent with the above policy, reprocessing and recycling facilities are operated in a way which minimizes the amount of liquid and solid waste produced : this has been a continuous COGEMA's strategy since the commissioning of latest plants at La Hague, UP3 in 1990 and UP2-800 in 1994.

2. REFERENCE VALUES : THE RELEASE AUTHORIZATIONS AND THE IMPACT STUDY

The whole industrial nuclear site of COGEMA-La Hague is submitted to compliance with the French and European Union Regulations, in particular with respect to radiological releases to the environment. Maximal authorized releases to the atmosphere and to the sea were prescribed in 1980 and are still in force. They are reported on Table 1.

Such values are based on acceptable exposure values for the neighboring population, according to ICPR recommendations and derived regulations. They also take into account the specific features of the site as concerns dispersion of radionuclides and local social habits.

Namely, an environmental impact study was performed to evaluate the maximal annual exposure of the neighbors, that would result from the authorized gaseous and liquid effluents.This study was validated by the French nuclear authority and by the European Union Commission.

The study followed general safety principles : in particular, it was grounded on very conservative assumptions.

Mainly,

- as concerns liquid releases to the sea, the critical group, determined as the local fishermen of Goury harbor, is mainly exposed by the ingestion of seafood ; for the purpose of evaluation, some concentration factors (from sea–water to seafood) and the assumed quantity of annually consumed seafood of local origin have been clearly overstated.

- as concerns releases to the atmosphere, the critical group has been determined as the local farmers, supposedly eating their garden production ; here again the ingestion contribution is calculated as dominant and conservative assumptions have been made about dispersion through the atmosphere, deposition into vegetation and animals and annually consumed quantities of local milk and vegetables.

Results are summarized on Table 2. The total annual exposure from liquid releases cannot be higher than 0.055 milliSievert, i.e. 1.2 % of the internationally established limit of 5 mSv. The same conclusion holds for the exposure from gaseous releases.

This very low estimated impact confirms that reasonable process options have been selected. It is justified to release and dilute C–14 and Kr–85 into the atmosphere. It is also justified to release and dilute tritium and iodine into the sea. Conversely, the extraction, concentration and containment of those radionuclides would mean unjustified occupational risks and additional volumes of waste.

3. ACTUAL RELEASES AND ACTUAL IMPACTS

Compared with the previously mentioned authorized values, the actually observed releases have steadily be maintained at a lower level. This is illustrated by Table 1 for 1994. According to the same impact model that has been used for authorized values, such real values would cause a lower than 0.01 milliSievert exposure to the neighboring people.

Moreover, a permanent monitoring of radioactivity content is ensured, in marine sediments and living species as well as in terrestrial vegetation and milk. It strongly suggests a still lower impact, for the following reasons :

- the measured radioactivity of human origin is generally lower than expected, even taking account of past and present actual releases ;

- the measured radioactivity of human origin has actually decreased in the marine environment since the reference year of 1965 ; preexistent activity was due to the falldowns of atmospheric weapon tests.

Improved instrumentation with lower thresholds of detection and more realistic assumptions on people's eating habits would be required, to test the model of impact against observed values.

Anyway, the most significant conclusion is the demonstrated negligible impact of the reprocessing activities at La Hague. This achievement is partially to associate with past R&D efforts on the process and on the plant management. The observed evolution from 1977 to 1994, as shown on Figures 2 to 5, clearly demonstrates that the released quantities are steadily reduced in spite of increased reprocessed quantities. Still more significant, the amount of release, divided by the amount of electricity (expressed as MW.year) produced from the reprocessed spent fuel, evidences a steep decline in time.

From 1987 on, the positive effect of modern plants is also apparent : the new liquid effluent processing unit in 1988 and the new reprocessing plant UP3 in 1989. From 1991 on similarly, the implementation of liquid effluent recycling within UP3 still reduces the amount of liquid releases as shown on Figures 4 and 5.

4. KEY TECHNICAL FACTORS OF IMPROVEMENT

The above mentioned achievements resulted from a steady policy of process optimization associated with targeted R&D programs. Three noteworthy factors of improvement will be described here.

4.1 The control of iodine release to the atmosphere

The minimization of iodine isotopes 131 and 129 release to the atmosphere is a key factor of public health protection. As an order of magnitude, the average spent fuel contains about 1 GBq of I129 per ton of uranium.
Considering an expected production of 1,600 t per year at La Hague, compliance with the authorized release of 111 GBq/year means that only 7 % at most of the content may be released through the stack.

In the UP3 plant, most of the iodine vaporizes during spent fuel shearing and dissolution. The whole gaseous stream from shearing and dissolution is processed in several steps (see Figure 6) :

- assisted desorption from the nitric solution,
- nitric acid condensation and recollection,
- scrubbing with water,
- scrubbing with aqueous soda,

- filtering in 3 steps, including 2 barriers of highly efficient filters and 1 specific iodine barrier.

Soda scrubbing ensures an abatement yield greater than 96 % for iodine. The subsequent iodine trap has been installed mainly as a safety added barrier for incidental situations. R&D investigations allowed to compare different techniques and to test the selected one during pilot experiments. It consists of zeolites, impregnated with a silver salt and offering a very high specific area for gas contact. Such traps can achieve a decontamination factor as high as 1000.

Other small sources of iodine release exist in the plant, because a small fraction remains in the liquid at the dissolver step and then slowly goes off from subsequent process steps, directly through the stack without processing. For illustration, the partition balance observed in 1992 is indicated on Figure 6.

Cogema has succeeded in maintaining soda scrubbing efficiency as high as possible, since it is more recommendable to drive iodine towards dilution into the sea than to produce more solid waste with zeolites.

4.2 The chemical treatment of liquid releases (Figure 7)

The chemical unit STE3, in operation since 1988, can process a liquid stream of 100,000 m3/year. The chemical process takes place in a series of stirred tanks into which specific reagents are introduced to precipitate each radionuclide : caesium, strontium, ruthenium, antimony and actinides mainly. The obtained active sludges, after settling, are conditioned into bitumen.

However, the efficiency of the process is increased by optimized implementation conditions. Three key factors have to be mentioned :

- the prior separation of residual organic liquid, coming from the solvent extraction units ;

- the steady pH and EFM monitoring during precipitation, which controls the rate of injection of chemical reagents ;

- the filtration step after flocculation of the final effluent, on a layer of diatoms, which stops all the suspended particles with a diameter greater than 25 microns.

Satisfactory performances for pH control and for filtration have been achieved on the basis of R&D experiments.

4.3 Improved liquid effluent management

The most important feedback from the operation of the UP3 plant is that the volume and the activity of low and medium level effluents was, from the beginning, significantly lower than estimated at the design stage. This comes from the very high performances of the extraction cycles, resulting in lower than expected activity levels for some of the effluents.

The improvements in liquid waste management are based on both a more sophisticated segregation of the waste according to their chemical and activity content, and the implementation of additional evaporation capacities in the plant. The effluent segregation allows to discharge the very low active streams to the sea whenever possible after filtration and monitoring, without increasing the activity released.

The remaining part of effluents to be considered are those from the analytical laboratory. These effluents traditionally include a mix of sample excess and chemical reagents which are not present in the main process. Moreover, the analytical effluents contain various levels of alpha and beta–gamma activity. The improvement of their management first relies upon a segregation within the analytical lines. According to their nature, they will be recycled to the plutonium purification cycles (alpha bearing effluents) or routed to the vitrification unit (beta–gamma effluents) whenever possible. On going R&D is dedicated to the following two improvements. :

- for the effluents containing unwanted chemicals, a special coprecipitation unit will be used to separate the alpha activity ;

- the amount of unwanted ions such as phosphates, sulfates, and chlorides is being significantly reduced through the use of alternative analytical methods.

With the start–up of these new units, which is ongoing, it will be possible to route practically all the activity concentrated to the existing vitrification facilities. Thus, the need for coprecipitation and bituminization treatments will practically disappear in normal operation for the intermediate and low level effluents from the UP3 and UP2–800 plants. The resulting small increment of activity conditioned in the glass will not significantly increase the volume of glass produced, but the overall volume of high level and long-lived waste will decrease below 1 m^3/t (see Figure 8).

It should be noted that improvements described above could be introduced without interruption of the production because the initial design accounted for further interventions or modifications.

5. DISCUSSION

We have shown how radioactive releases have been minimized at La Hague. The implemented policy has always applied the ALARA optimization principle. Generally speaking, a reduction of release should not bring about new volumetions or hazardous solid waste ; the plant impact should be minimized globally.

As a first case, a lot of work has been done on the capture and immobilization of krypton. A first conclusion was the complexity of the successive treatment steps (removal of moisture, oxygen and nitrogen oxides) requiring, inter alia, the use of hydrogen gas. The separation would finally result in a very concentrated form of absorbed krypton on zeolites, to be stored for decay during a relatively long period, since the half–life is 10.5 years. The second conclusion then is the greater potential hazard induced locally by such a storage of large quantities, compared with the global impact from dilution into the earth atmosphere. The world reprocessing capacity would have to be drastically increased, to justify krypton capture.

A second case to discuss is the dilution of I129 into the sea. Even under conservative assumptions, it induces very low doses to the local people (less than 1 microSievert per year). An alternative would be to trap I129 as a solid waste, to be buried in a geological repository. But here again, such a concentrated waste would become a potential hazard at long, since I129 is likely to have a longer duration of life than encapsulating materials and so to escape and return to the local biosphere in the long run.

Much more convincing is a process improvement which simultaneously reduces releases to the environment and waste volumes. This is the case of the on–going extended recycling of effluents which is about to suppress any bituminized solid waste, while reducing releases to the sea.

6. FUTURE TRENDS

Finally, the expected evolution of releases from La Hague could be summarized as follows :

- implemented abatement processes ensure that releases of highest impact are not coupled with increased production. This is clearly demonstrated for liquid releases, gaseous aerosols and gaseous halogens ;

- releases of uncontrolled radionuclides (tritium, C–14 and Kr–85) will grow along with production, but within the frame of authorizations which are clearly preserving environmental quality. These radionuclides are definitely innocuous after dilution into the earth atmosphere.

7. CONCLUSION

In the field of environmental radiological protection, process R&D has definitely provided efficient means to minimize potentially hazardous releases. Complementary gains are possible by operating optimizations, but they should be measured globally and they will result from a systemic analysis, rather than localized advanced performances. Current proprietary R&D programs are oriented by this general approach.

Table 1

Assessment of the releases at La Hague in 1994

Gaseous releases

	Authorization	% / Authorization 1994
Tritium	2 200 TBq	2.5 %
Halogens	110 GBq	20.1 %
Other gases	480 000 TBq	37.4 %
Aerosols	74 GBq	0.016 %

Liquid releases

	Authorization	% / Authorization 1994
Tritium	37,000 TBq	21.9 %
Beta (other than tritium)	1,700 TBq	4.1 %
of which 137 Cs + 90 Sr	220 TBq	11.9 %
Alpha	1.7 TBq	5.7 %

Table 2

Total annual exposure of the local critical groups

From releases to the sea	:	55 microsievert (90 % internal)
From releases to the atmosphere	:	60 microsievert (80 % internal)

Figure 1

Annual average electricity equivalent reprocessed at Cogema – La Hague

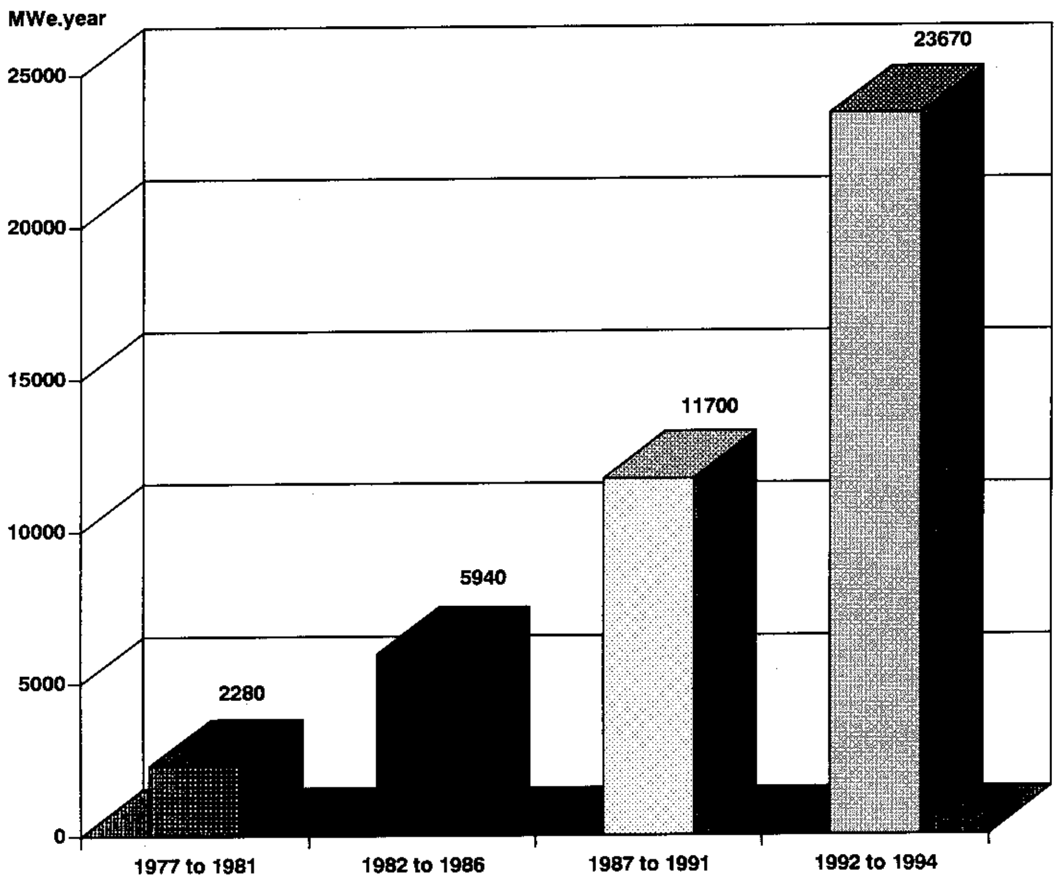
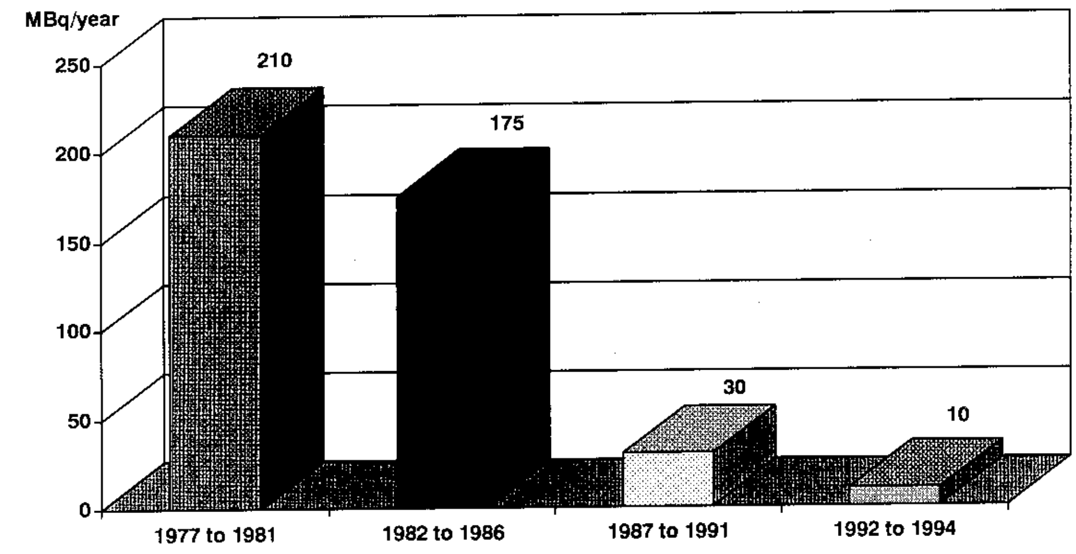


Figure 2

Gaseous releases : aerosols at Cogema – La Hague

Annual average activity



Ratio released activity/produced electricity

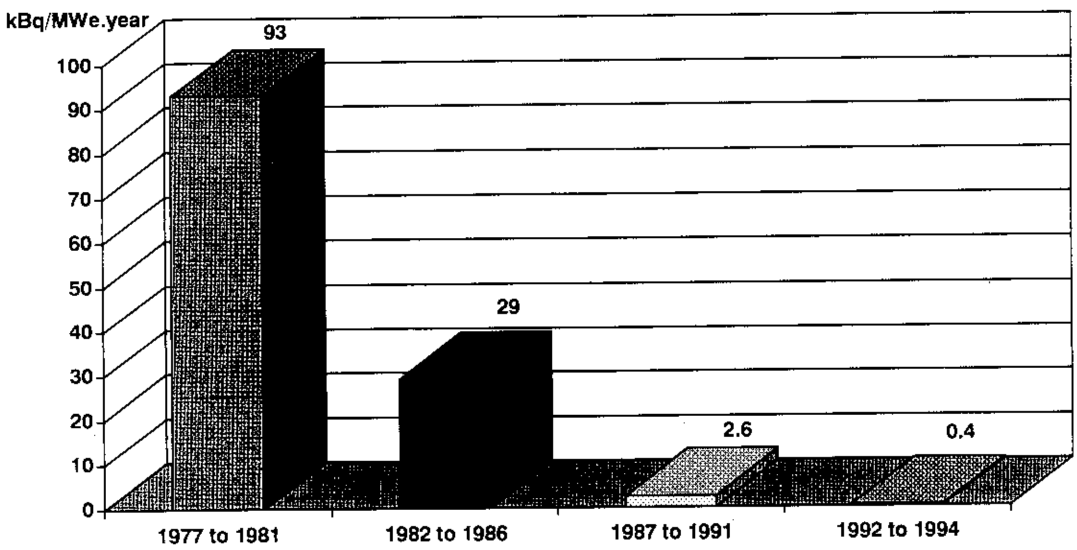
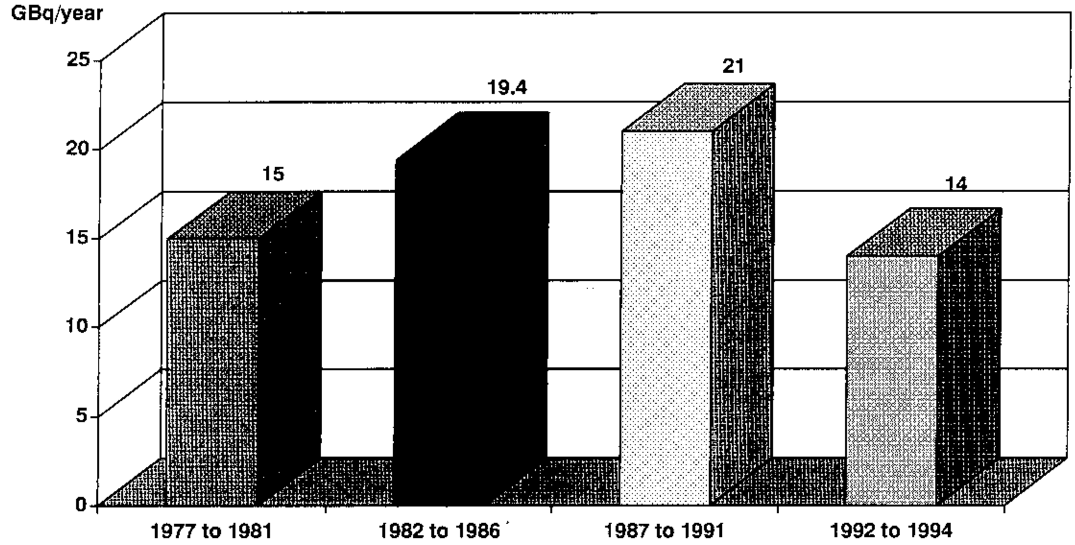


Figure 3

Gaseous releases : halogens at Cogema – La Hague

Annual average activity



Ratio released activity/produced electricity

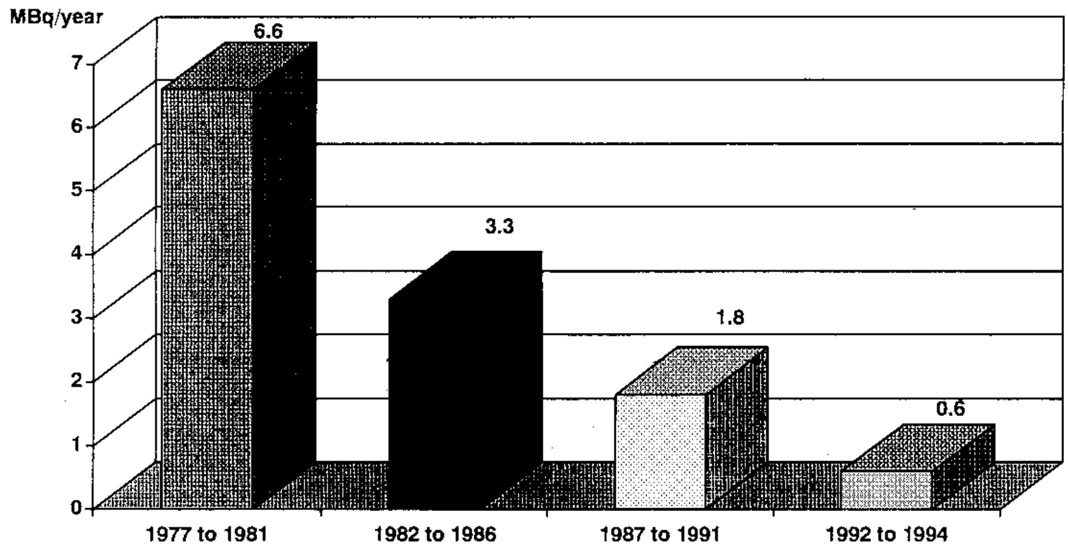
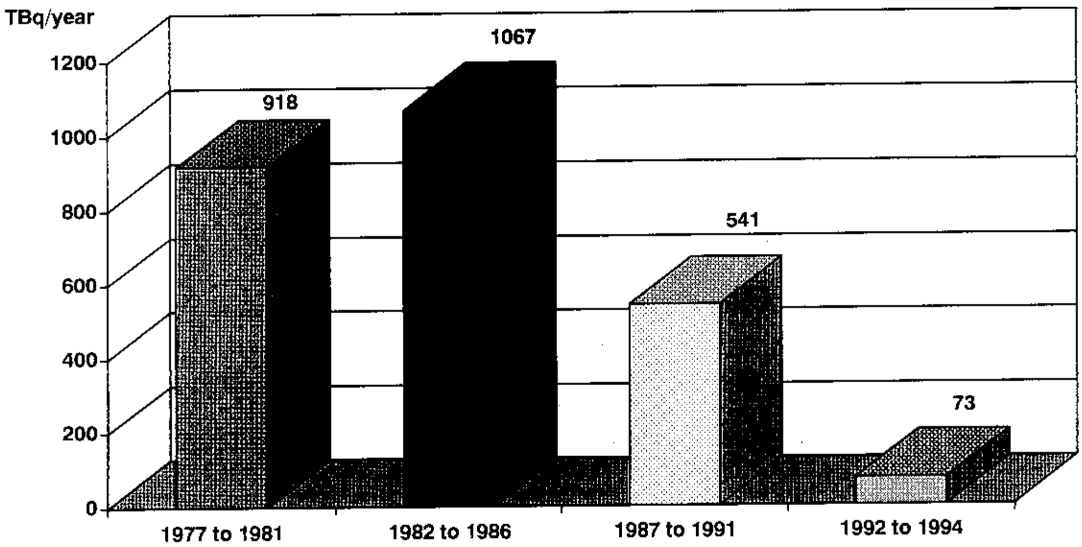


Figure 4

Liquid releases : $\beta\gamma$activity without H_3 at Cogema – La Hague

Annual average activity



Ratio released activity/produced electricity

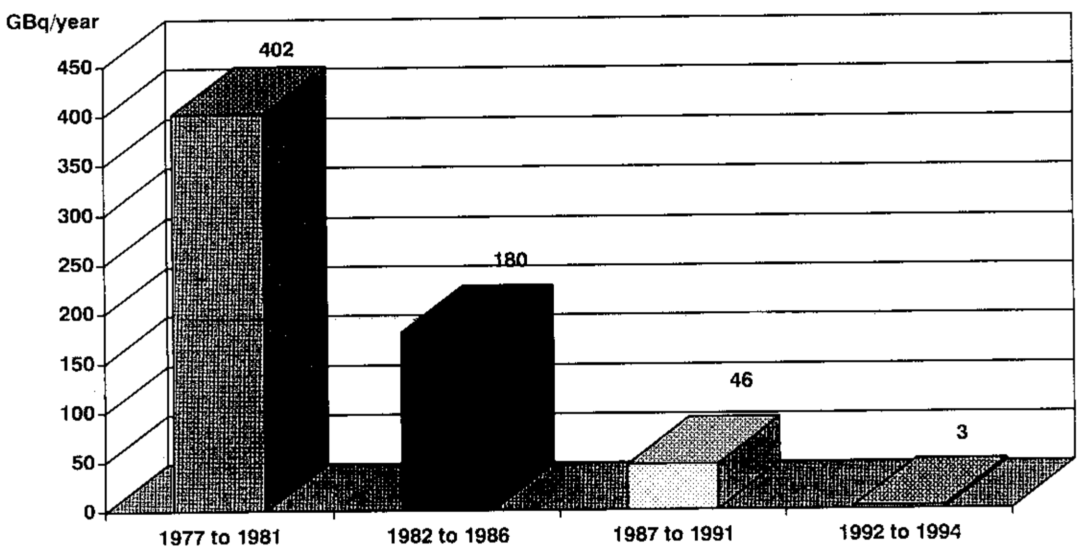
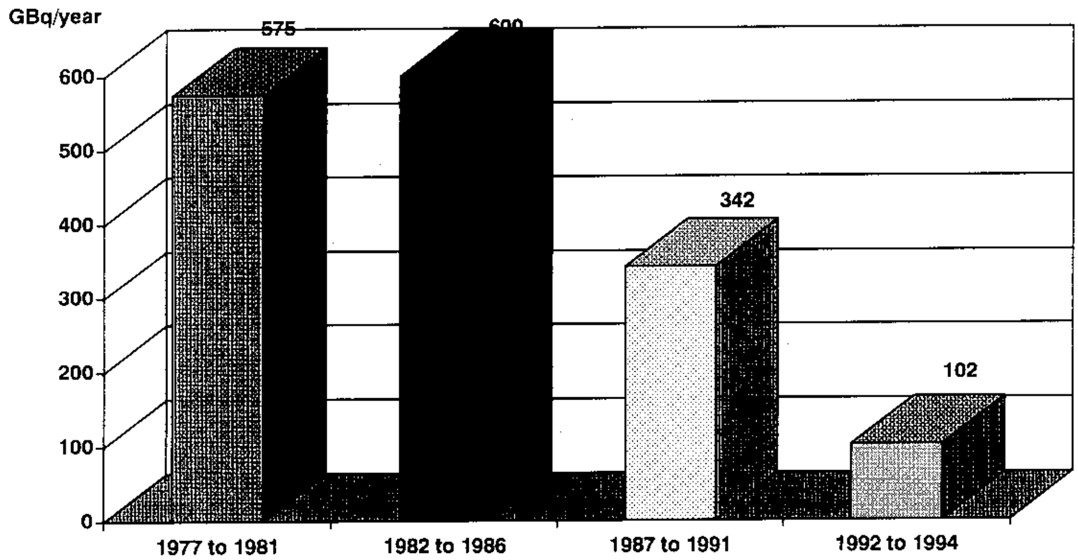


Figure 5

Liquid releases : α activity at Cogema – La Hague

Annual average activity



Ratio released activity/produced electricity

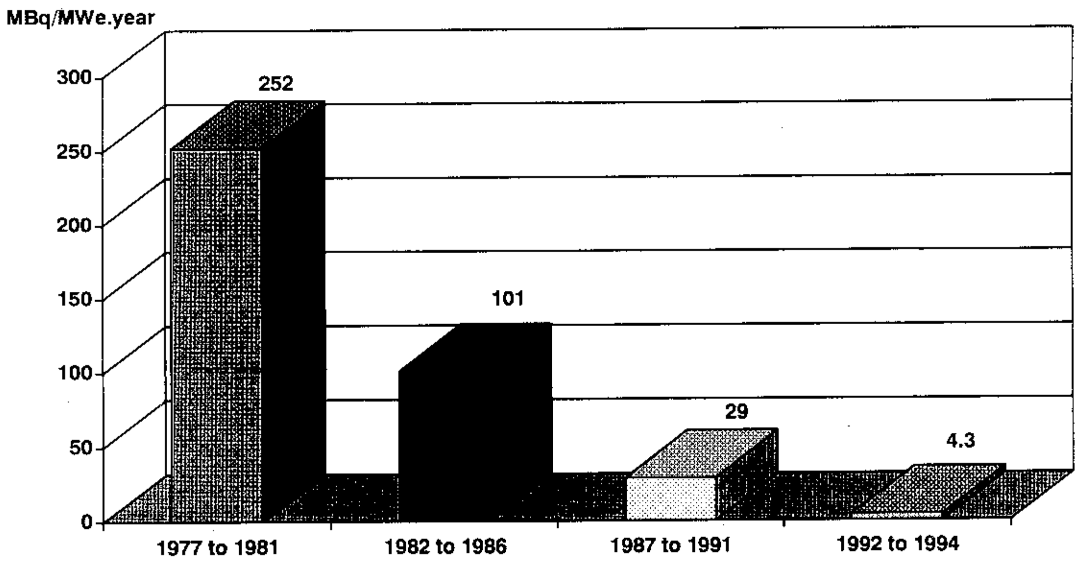


Figure 6

Iodine extraction from exhaust gases

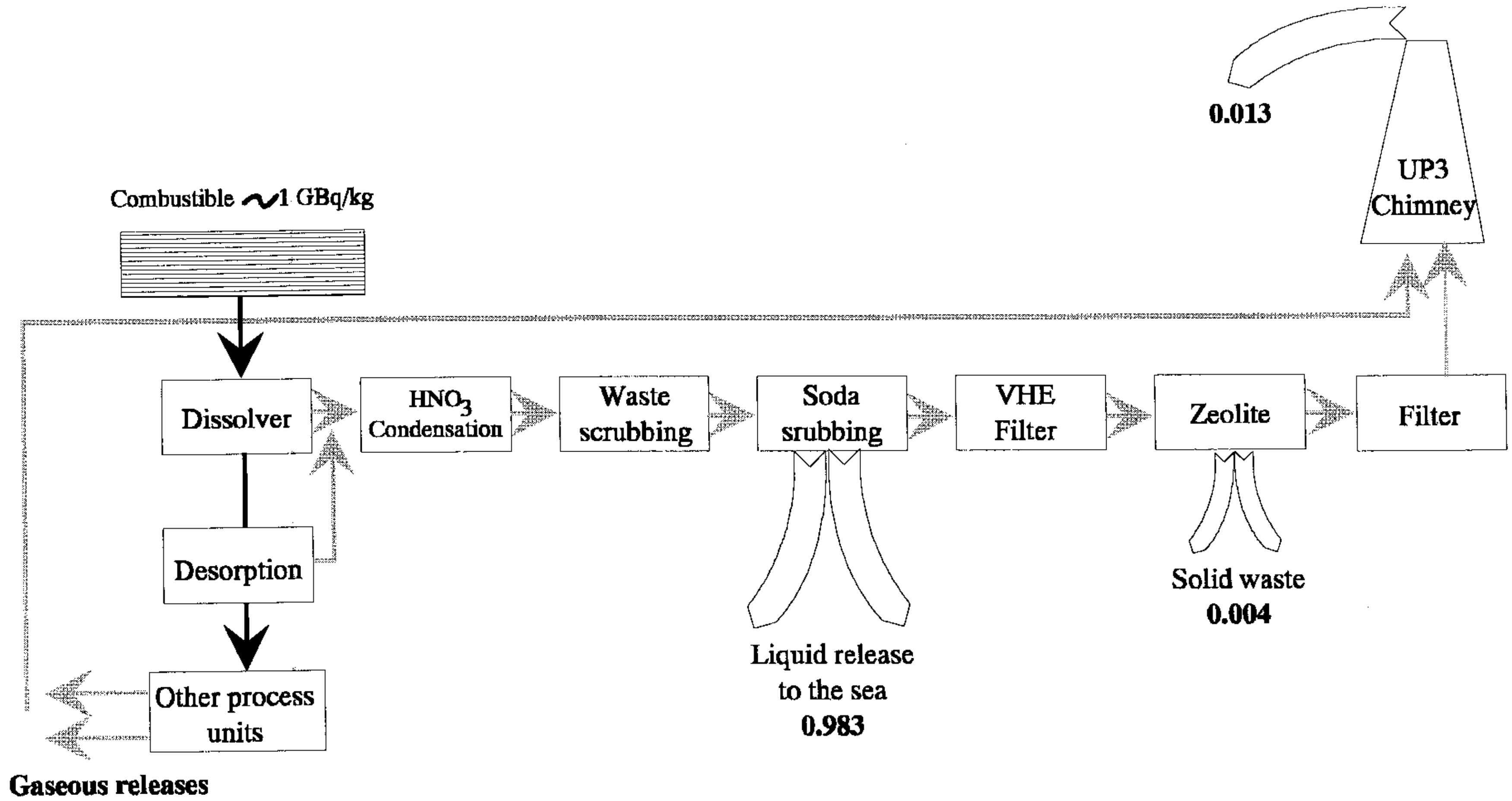


Figure 7

Simplified diagram of the liquid effluent treatment

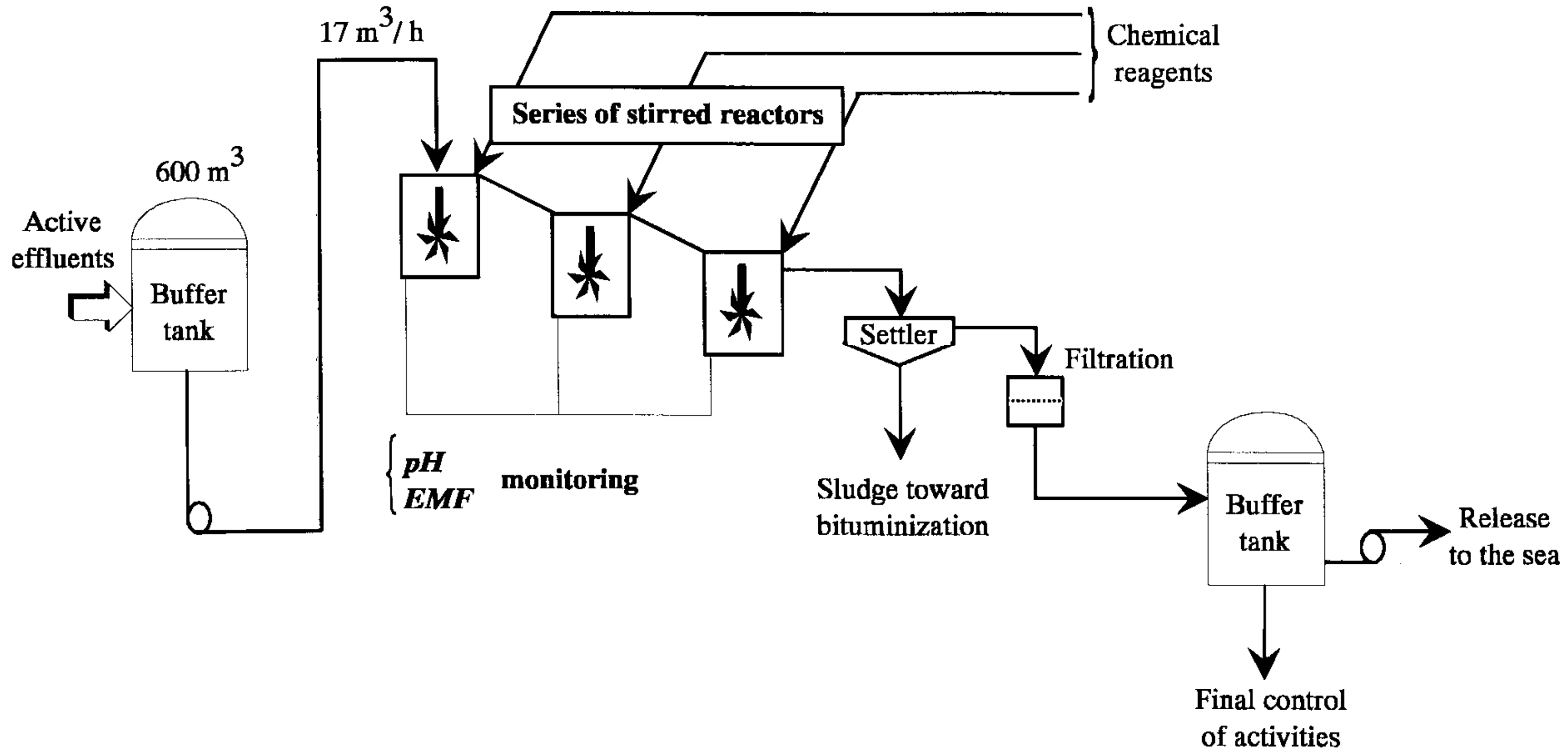
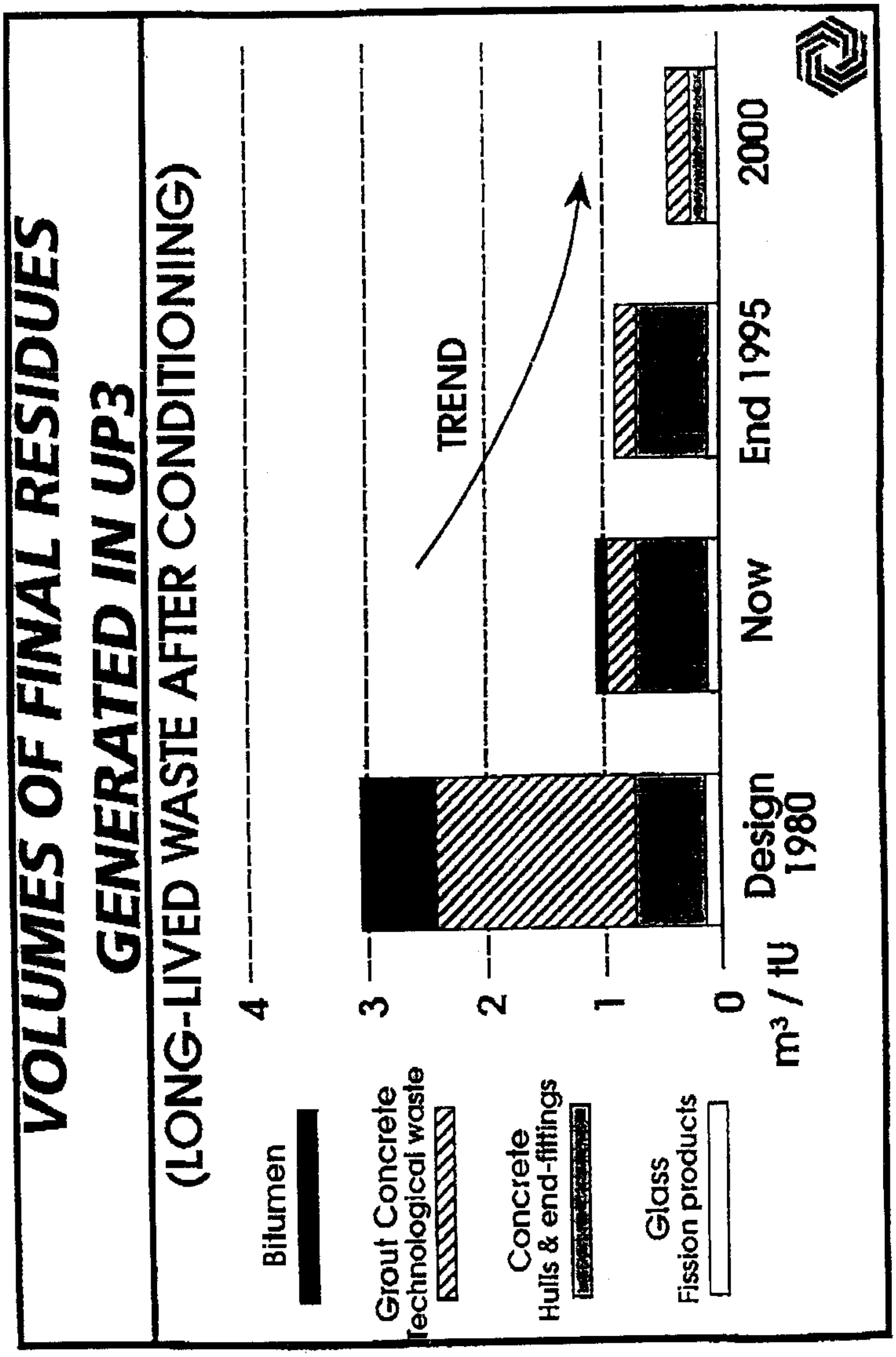


FIGURE 8


VOLUMES OF FINAL RESIDUES
GENERATED IN UP3
(LONG-LIVED WASTE AFTER CONDITIONING)
Bitumen
Grout Concrete
Technological waste
Concrete
Hulls & end-fittings
Glass
Fission products
4
3
2
1
0
m³ / tU
TREND
Design
1980
Now
End 1995
2000

THE MINIMIZATION OF RADIOACTIVE RELEASES TO THE SEA FROM THE TOKAI REPROCESSING PLANT

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Abstract

The Tokai Reprocessing Plant(TRP) started hot operation in September 1977. The total amount of about 790tU of spent fuel, generated in Japan, has been successfully reprocessed as of December 1994. Low-level liquid wastes have been treated safely with the low-level waste treatment process.

The design of TRP was based on foreign technology. In the early stage of designing, the radioactivity released to the sea was estimated at approximately 2.6 TBq/day (70 Ci/day) for beta activity (except for tritium). Later, PNC added an evaporator to the process to reduce the level down to 1/100, i.e. 9.6 TBq/year (260 Ci/year) or 2.6×10^{-2} TBq/day (0.7Ci/day). In addition, under the supervision of the government, PNC started R & D to further decrease the radioactivity released to the sea in terms of ALARA. Aiming at reducing the activity from 9.6 TBq/year (260 Ci/year) to 1/10 of that value (i.e. 26 Ci/year), the release reduction technology development facility was added. This facility was incorporated into the low-level waste treatment process in 1980, before starting the regular operation of TRP.

Since the fuel reprocessing commenced, total radioactivity discharged to the sea has been 1.9×10^{-2} TBq (0.51 Ci) for beta activity, as of December 1994. Before incorporating the release reduction technology development facility, the yearly level was $3.7\times10^{-3}\sim7.4\times10^{-3}$ TBq ($0.1\sim0.2$ Ci). After incorporation of the facility, radioactivity released to the sea was greatly decreased to non-detection levels in recent years, in spite of increasing annual reprocessing amounts. Although serious equipment failures have occurred such as the acid recovery evaporator and the dissolvers, there was no influence on radioactivity released to the sea.

1. INTRODUCTION

1.1. Overview of Tokai Reprocessing Plant.

PNC started on the reprocessing project in 1956 when Atomic Energy Commission (AEC) of Japan decided that reprocessing of spent fuel and treatment of radioactive waste should mainly be done by Atomic Fuel Corporation (AFC, the predecessor of PNC). In 1959, AEC formed Advisory Committee for Reprocessing within AEC to formulate a guideline for development of reprocessing technology. Along with AEC's and Advisory Committee for Reprocessing decision, AFC began the preparation for construction of reprocessing plant in Tokai-mura, about 100km northeast of Tokyo.

After studying the situation of foreign reprocessing plants, Advisory Committee for Reprocessing recommended that it was desirable to construct a practical size plant of 0.7-1 t/day based on a foreign technology, in 1962. In 1963, the AFC entered contracts of preliminary design of plant with foreign companies, and detailed design was carried out by SGN from 1966. In 1967, PNC succeeds to reprocessing project from AFC. Since 1968 and in parallel with the ongoing detailed design, the governmental licensing procedure had been followed and permission for plant construction was gained in 1970.

As shown in Table Ⅰ, plant construction was started on 1971 and was completed on 1974. In this construction period, blank test was made in parallel. Chemical test and uranium test (using unirradiated uranium) were proceeded step by step. The active test was started at September 1977, and the prior inspection for operation was conducted. PNC got the license for operation in 1980, and operation started on 1981. Recently, the scheduled long period of shut-down of plant operation was set to replace and improve the plant equipments, to prevent sudden equipment failure and to increase reprocessed amounts.

Up to the end of 1994, the total amount of reprocessed fuel from LWRs and ATR (Advanced Thermal Reactor using heavy water as the moderator) "Fugen" was about 790 tons, with a maximum burn-up of 35000MWD/tU. Total amounts of recovered plutonium nitrate as final product reached about 5.4 tons up to December 1994. Most of recovered Pu has already been sent to Plutonium Conversion Development Facility for use at the ATR "Fugen", the experimental FBR "Joyo", and proto-type FBR "Monju"[1]. PNC came up with some difficulties through this reprocessing project. This paper deals with the experience of the low-level liquid waste treatment.

1.2. Difficulty in the Low-Level Liquid Waste Treatment.

Difficulty was occurred because of social and environment change[2]. This difficulty was beyond what was considered at initial designing of plant. The management of radioactive waste in Japan is important problems, because not only very densely populated but also people depend much upon fish and shells for their important resources of protein. So, Japanese interests in the sea are immeasurable. Construction permit authorization was given by government in 1970 and construction work was started on 1971, but demand for improvement for environment safety continued even after construction started.

2. Description of the Low-Level Liquid Waste Treatment Process

2.1. Origin and Classification of Liquid Wastes.

The liquid wastes generated from the reprocessing plant are classified into high-, intermedium-, and low-level (HLLW, ILLW, LLLW). The origin of liquid waste are shown in Table Ⅱ.

HLLWs and ILLWS are treated in the separations and purification plant which usually called MP (Main Plant), whereas LLLWs are transferred to low-level waste treatment facility, which is usually called AAF (Auxiliary Active Facility), through underground pipelines after routinely checking up activity and chemical properties.

2.2. Initial Process.

At the initial design, TRP was provided with LLLW treatment facility with AAF only.

AAF is provided with one unit evaporator, one unit flocculater, and one unit sand filter. This evaporator is self-thermocompressing natural-circulation type with capacity of $50m^3$/day, and provided with a cyclone and a wire mesh demister to remove mist, which is called 1st.-evaporator now. The flocculater has capacity of $120m^3$/day, which called chemical treatment process. Coprecipitation was made by adding of calcium carbonate and ferric hydroxide.

LLLWs were treated with some of above equipments depend on their radioactive concentrations and chemical properties. This initial process is simple, as shown in Fig.1. Higher-LLLW, whose activity concentration is $370kBq/cm^3$ to $37Bq/cm^3$, is subjected to one-stage evaporation. Whereas lower-LLLW, whose activity concentration is less then $37Bq/cm^3$, is coprecipitated and filtrated. The distillate from acid recovery process, however, was discharged without any treatment. The resultant concentrate and chemical sludge are stored in the exclusive storage vessels.

2.3. Modification of the Process.

2.3.1. Oil removal from effluent.

In 1976, oil contamination limit of sea discharge effluent was tighten because of revised chemical pollution regulation. PNC finds that the oil concentrations in distillate from the acid recovery process in MP and 1st.-evaporator were over alternate limit sometimes. To solve this problem, PNC entered design for oil removal facility and chemical treatment process was modified temporary by adding the active carbon to remove entrained TBP, as shown in Fig.2. The addition of active carbon was very effective in reducing the oil discharge.

2.3.2. Addition of Evaporator.

Although the initial regulation value on the sea-discharge was just 37GBq/day (except for tritium), demand for improvement in environment safety enhanced during the construction period. Fisherman's associations and related organizations entered suit against construction work. To avoid this problem, PNC modified the low-level liquid (LLLW) waste treatment process by constructing another facility. This facility called Second LLLW evaporation facility (usually called E-facility).

E-facility provides one unit of natural-vertical-circulation evaporator with capacity of $90m^3$/day. This evaporator called 2nd-evaporator. 2nd-evaporator treated the distillate from 1st.-evaporator. The evaporation technique is very effective to produce the sufficient low-level distillate and reduces the radioactivity in the sea-discharge effluent. E-facility was designed and constructed by domestic companies, and construction was completed early 1975, before beginning of active test. This evaporation facility was incorporated with LLLW treatment process to modify the process as shown in Fig.3.

2.3.3. Further Addition of Evaporator.

In addition to the above, PNC has been made effort to reduce the radioactivity discharged to environment from Tokai Reprocessing Plant (TRP) along with "ALARA" philosophy. PNC decided to construct 3rd-evaporation facility (usually called Z-facility in connection with Zero discharge).

Z-facility contained one unit evaporator. This 3rd-evaporator is same type as 2nd-evaporator but it capacity is $210m^3$/day. The construction of this new facility start on 1977 and completed on 1979. This facility was designed and constructed also by domestic companies.

In the early time of the active test, PNC came up with difficulty of high alpha activity in the distillate from acid recovery process which exceed the regulation value for sea discharge. As a result of provisional measurement, PNC was obliged to modify the process again to treat a part of the distillate from the acid recovery process and a part of lower-LLLW by 2nd-evaporator, but it had no capacity to treat all liquids. So, PNC intended to use 3rd-evaporator for operation routinely. PNC had made the tests using actual liquid waste during the active test and the results showed sufficient decontamination factors. The modification of this process is carried out as shown in Fig.4. 3rd-evaporator has enough capacity to treat the produced waste. So, the 2nd-evaporator retired from the service and became supplemental evaporator now.

2.3.4. Effluent Clean-up by Charcoal Adsorption Column.

In order to protect the surrounding seawater from chemical pollutants due to entrained TBP, the charcoal adsorption columns installed before sea-discharge process, instead of the modified chemical treatment. This new facility has charcoal adsorption columns, and which called the Oil Removal facility (usually called C-facility). C-facility was incorporated into the process in 1979. The final process for LLLWs is shown in Fig.5. Higher-LLLWs are subjected to two-stage evaporation, whereas low-LLLW is coprecipitated or evaporated in one stage, and the distillate from the acid recovery process (which was not treated at the initial process) is also subjected to one-stage. To reduce radioactivity discharge, part of lower LLLW is treated by evaporator. Finally, decontaminated liquid wastes are de-oiled in charcoal adsorption columns. LLLWs treated as above are discharged through sea-discharge pipeline after confirming the radioactivities and the chemical pollutants in the effluent.

3. Results.

LLLWs are decontaminated by evaporators or flocculater sufficiently below the discharge limits (2.96×10^{-2}Bq/cm^3 of alpha activity and 12.2Bq/cm^3 of beta activity) and are further de-oiled below 5.0ppm before being discharge into the Pacific Ocean, about 3.7km apart from seaside. As shown in Fig.6 and Fig.7, discharged radioactivity to the sea is remarkably decrease since 3rd-evaporator has been incorporated in LLLW treatment process since 1980. Up to December 1994, total discharged radioactivity is only 4.6×10^{-4}TBq for alpha, 1.9×10^{-2}TBq for beta (except for Tritium). On the other hand, concentrate have been fixed in bitumen since October 1982.

4. Another Improvements.

4.1. Reduction of iodine discharge.

At the beginning of the active operation, we had the problem of that how to control the iodine released into the environment. In order to solve this problem, the measurements of iodine in the gaseous and liquid streams have been made through the active operation.

From the result of measurements, it was estimated that more than 99% of iodine in the spent fuel was released into off-gas circuit during the dissolution. Most of the iodine released to off-gas circuit was resorbed at scrubber to liquid streams (higher-LLLW) which was transferred to AAF and then subjected to two-stage evaporation[3]. Methods for decreasing the volatility of iodine include adjusting of the pH of solution adding NaOH to produce new conditions under which the species is non-volatile.

It is desirable to have as high volume reduction as possible to minimize the size and of concentrate vessel and reduce the burden on the bituminization process. PNC has made the test for reducing iodine discharge by the alkaline process. Test results indicated the technical feasibility of reducing iodine discharge sufficiently by adaptation the second stage only. Fig.8 shows that radioactive iodine discharged into the sea has been considerably reduced by adaptation of the alkaline process. The amount of radioactive iodine released into the sea is currently about 1% or less than of estimated inventory in the spent fuel.

4.2. Prevent the Foaming.

Foaming is a major problem associated with evaporation. A small amount of entrainment can reduce the decontamination factor to an unsatisfactorily low. Once foaming occurs, the decontamination factor decreases markedly due to the rise in liquid level, which causes an increase of entrainment and, in the worst case, carry-over of solution into the distillate. The ways of prevent foaming employed are improvement of evaporation control and ITV-monitoring.

5. CONCLUSION

The design of Tokai Reprocessing Plant was based on a foreign technology. However, social environment is different in each country and it has a great influence upon waste management. Accordingly, we modified the LLLW treatment process by our own technology, in order to fit and fix the Tokai Reprocessing Plant operation to the state of affairs in Japan.

This modification reduced radioactivity being discharge to the sea over many years, thereby satisfied the demands of environment safety. These results have been effective in obtaining the public acceptance for fuel reprocessing in Japan.

Reference

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2. Y.Nojima, et al. : "Operational Experience in the Low Level Liquid Waste Treatment at Tokai Plant", Proc. of Fuel Reprocessing and Waste Management, Vol.1, p.505-515, Jackson, 1984

3. G.Fukuda, K.Matsumoto and K.Miyahara : "Experience and Projects Concerning Treatment, Conditioning and Storage of All Radioactive Waste from Tokai Reprocessing Plant", IAEA-CN-43/131, Vol.2, p.279-292, Seattle, 1983

Table I Operation Schedule on Tokai Reprocessing Plant

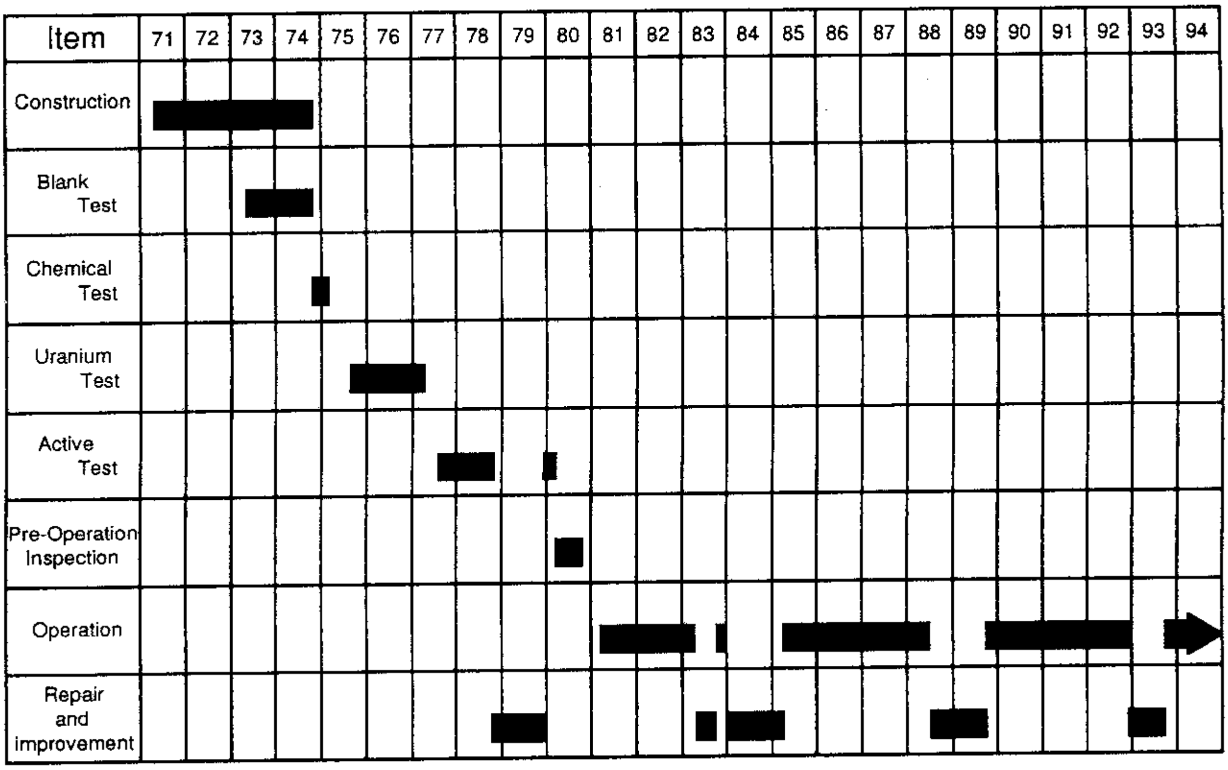


Table II Classification of Liquid Waste

High-level liquid waste	1st cycle effluent 1st cycle solvent wash effluent Concentrate from acid recovery process
Intermedium-level liquid waste	2nd and 3rd Uranium and Plutonium cycle Aqueous effluents DOG and HLLW vent condensates Plutonium evaporator condensate
Low-level liquid waste	Off-gas scrubbing effluents Storage pool cleanup system backwash Solvent wash effluent Floor drain Laundry effluent Distillate from acid recovery process

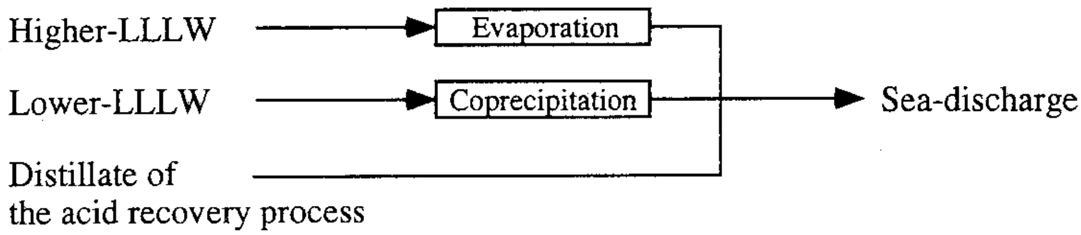


Fig.1 Initial LLLW Treatment Process

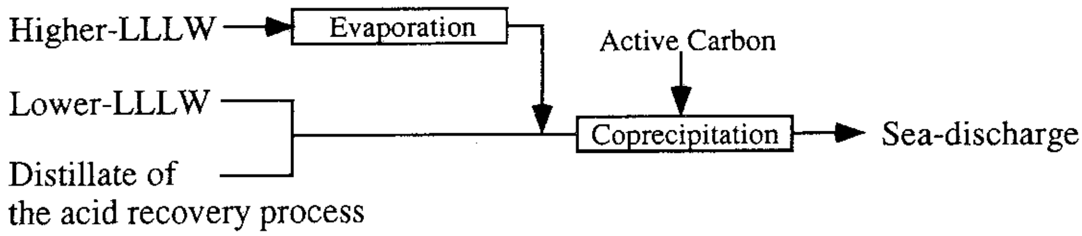


Fig.2 Modified Process

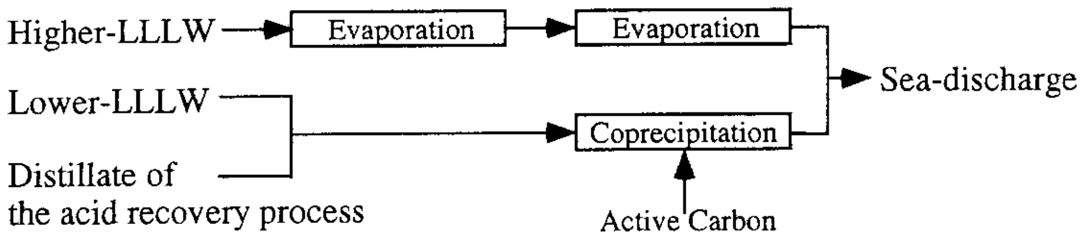


Fig.3 2nd-modified Process

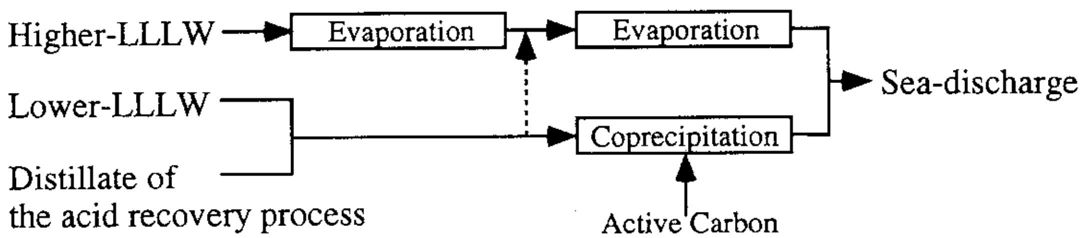


Fig.4 3rd-modified Process

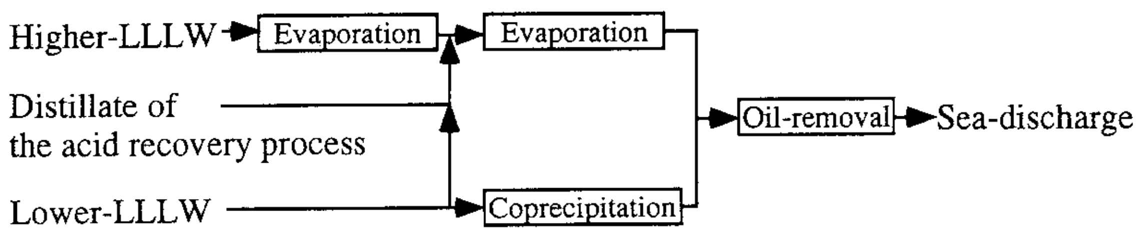


Fig.5 Final modified Process

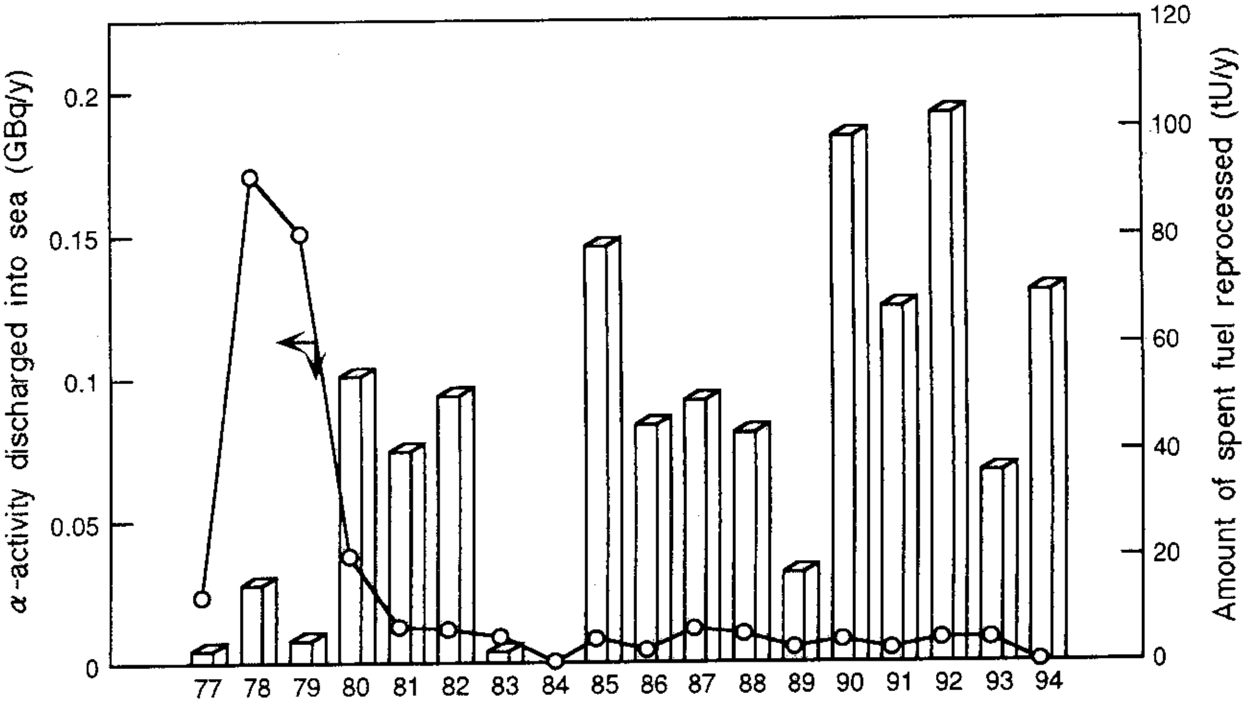


Fig.6 Discharged α-activity to Sea

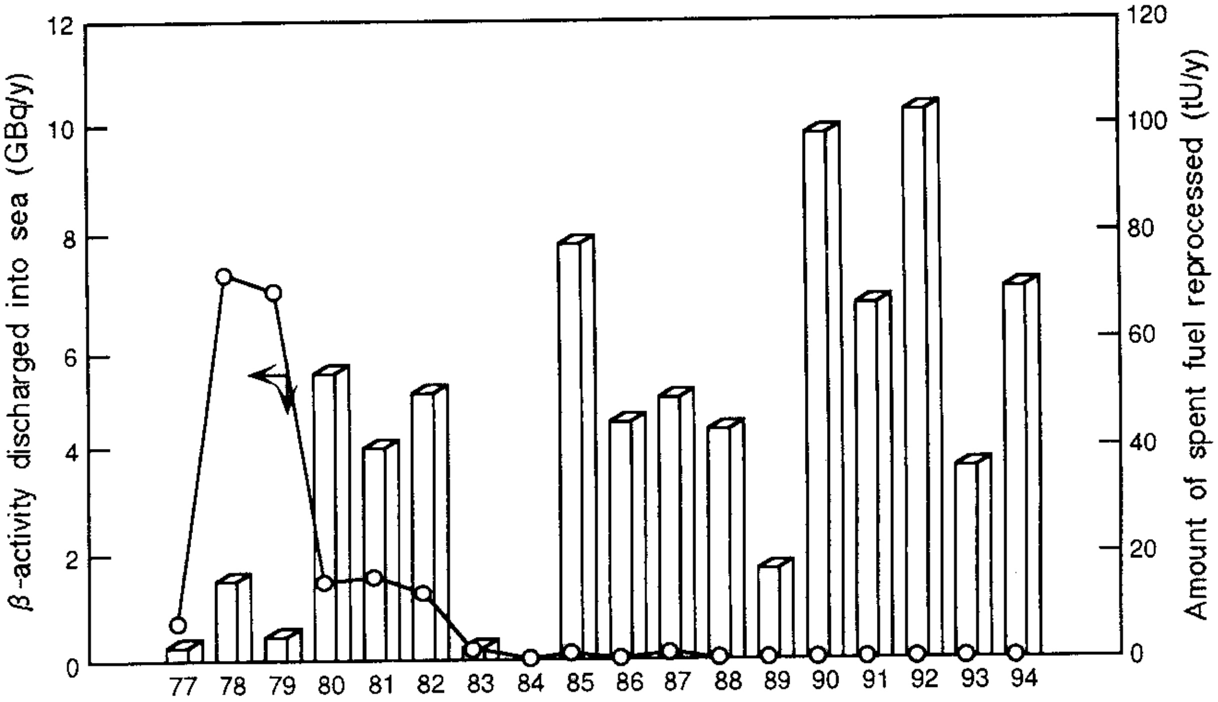


Fig.7 Discharged β-activity to Sea

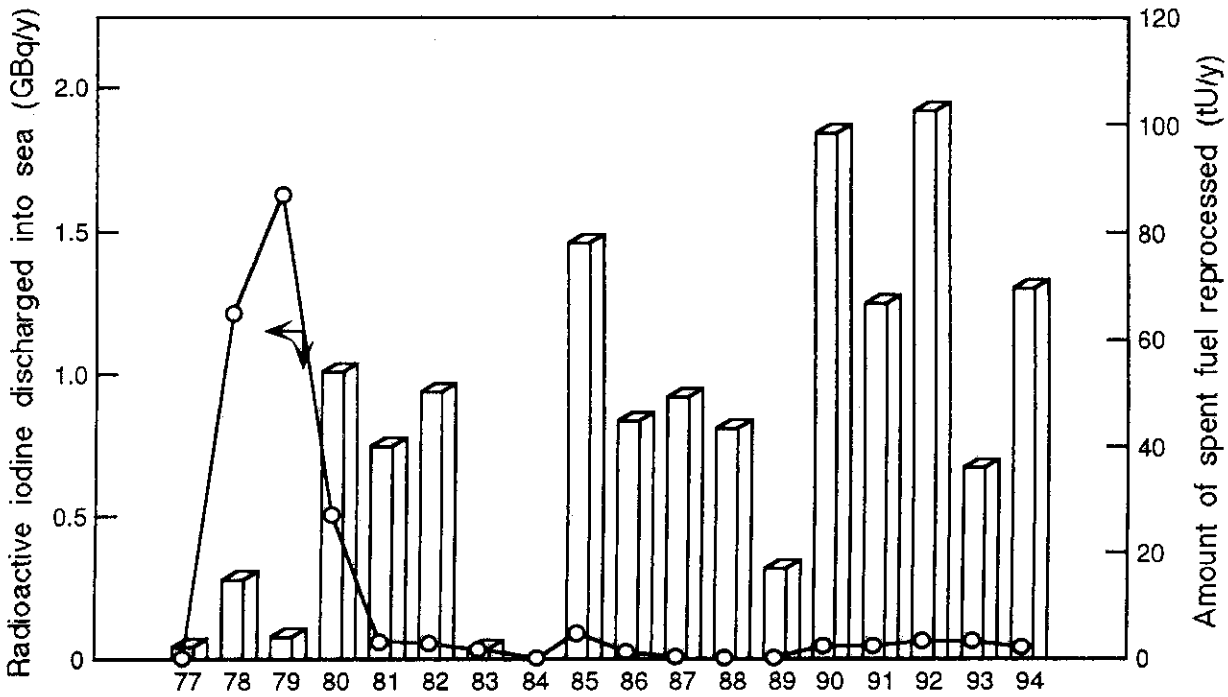


Fig.8 Discharged Radioactive Iodine to Sea

Tuesday, October 17th, 1995

Panel Discussion:

Strategies for Safety Research in the Field of Nuclear Fuel Cycle and Effective Utilization of NUCEF

Cochairs
T.Tsujino(JAERI, Japan),
M.L.Brown(OECD/NEA-FCS)

Panelists
J.P.Mercier (IPSN, France)
P.Watson (HSE/NII,UK)
C.V.Parks (ORNL, USA)
K.Hirose (STA, Japan)
M.Kanamori (PNC, Japan)
S.Matsumoto (Saitama Univ., Japan)
M.Maeda (JAERI, Japan)

Summary of Panel Discussion, addressed by the Chairman.

(1)Through various safety researches with systematic program and safety demonstration tests,

-Safety of fuel cycle facilities are at present maintained at high level. And feasibility of severe accident to release large amounts of radioactivities is very low at the current refined technology. Safety researches are further expected to improve safety quality, to clarify the safety margin for the reasonable regulation and efficient plant design or operation to bring economical merit.

-Safety researches are also important for the new and advanced technology like MOX recycle in thermal reactors toward recycling of plutonium in FBR, future actinide recycling, advanced waste conditioning to reduce volume and storage, transuranium partitioning, which are under development and going to be studied.

(2)Important area for the safety researches are;

-Criticality safety and shielding for MOX fuel and treatment of TRU.

-Confinement safety including release of radioactivity from plants and contamination transfer in an incident.

-Fire and explosions safety.

-Safe operation and maintenance: in-service inspection including corrosion.

-Handling problem for the reprocessing of MOX fuel and high burnup fuel, future high actinide containing waste.

In addition, long-time intermediate storage of spent fuel must be considered all over the world.

(3)Other important issues raised with respect to the utilization of results of safety research are;

-Results should be opened and explained using understandable form, for example publishing handbook or database, for the public acceptance or public understanding.

-Results are expected to be smoothly transferred to industrial parts to contribute to effective design and rational licensing. They are bringing effective safety operation and also economical merit.

-Due to safety characteristics of fuel cycle facility, comparing with the nuclear power plants, especially fuel treatment facility like reprocessing, potential hazard to bring accident is very low. When we consider postulated large accident, definition of beyond design basis event is expected to be clarified. And because the fuel cycle facilities are very complicated and its large area of installation, the number of small incident may be more. We must be careful in the treatment and handling of such small incidents. And they are deeply analyzed by the FINAS group of OECD/NEA FCS-WG.

(4)For these safety researches, utilization of NUCEF is effective to collect experimental data including comparison with analytical computed results.

-Expectation for not only domestic cooperation but also international cooperation is recognized.

-NUCEF also could be utilized, although it is at the beginning stage, to be the safety research center for fuel cycle, in the future. This symposium is kick-off and the beginning place for information exchange and discussions.

Tuesday, October 17th, 1995

Concluding Remarks

Concluding Remarks, by T. TSUJINO

Finally

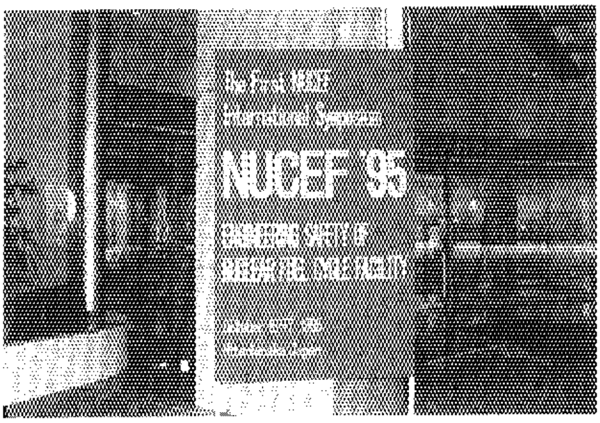
Good evening, ladies and gentlemen

(1) It is a great pleasure for me to say that the NUCEF'95 that successfully finished with interesting presentations and also active discussions. We also appreciated very much for so many participants unexpected up to 270, especially for foreigners who came to Japan after a long flight.

(2) We sincerely hope that the NUCEF will generate good research results as expected for not only rational safety assessment, but also for safety improvements to bring any economical merits. And also, the NUCEF will become experimental center for safety researches in fuel cycle under the domestic and international cooperation.

(3) The NUCEF'95 will at first be followed by small expert meetings which will be held every one or two year and the next international symposium; NUCEF'98 will be held upon the another subjects entitled; TRU separation and TRU waste management, which is connected with BECKY installations in NUCEF.

(4) Now, I would like to declare the closing of this symposium and thankyou very much again for your participation and cooperation. Sayonara, Good bye, Au revoir, Auf vieder shen.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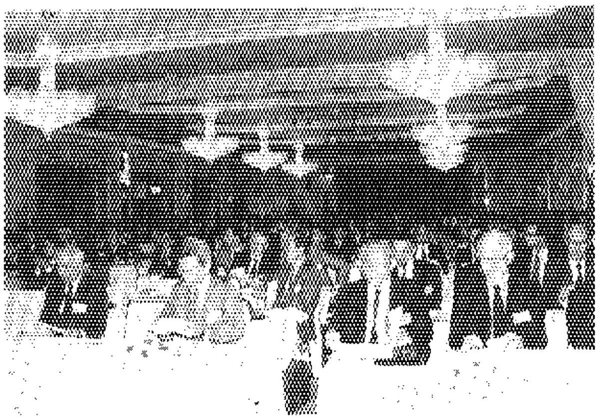

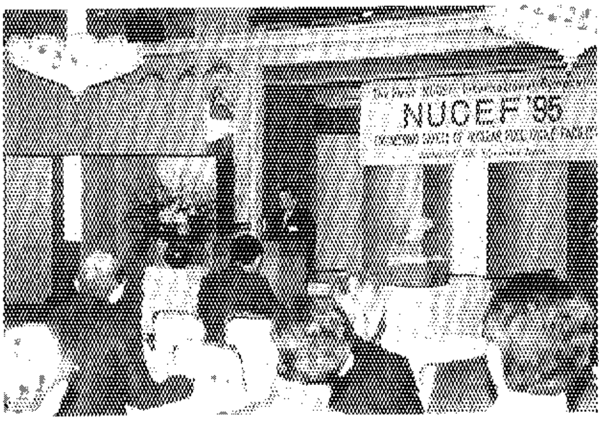
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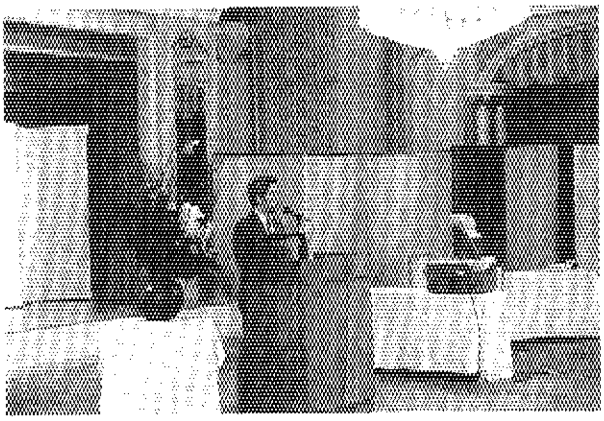
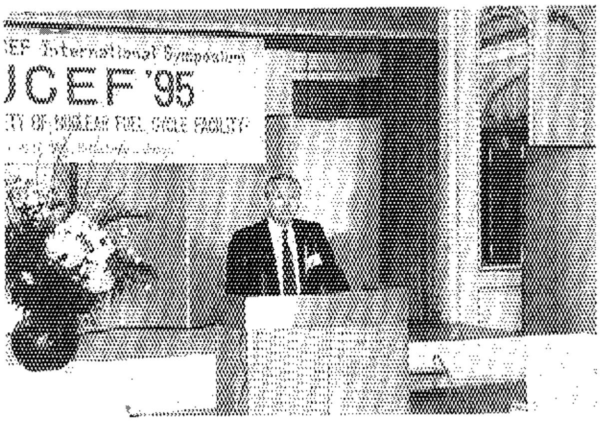
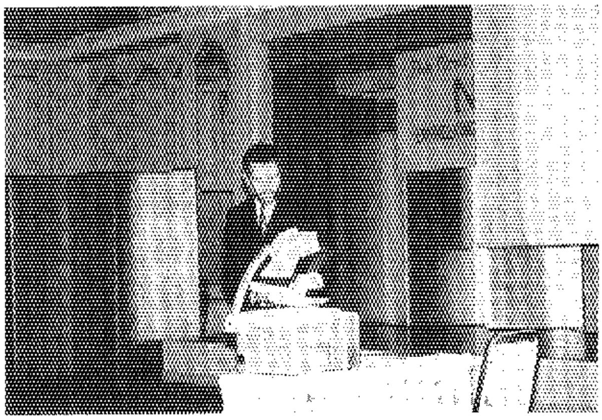
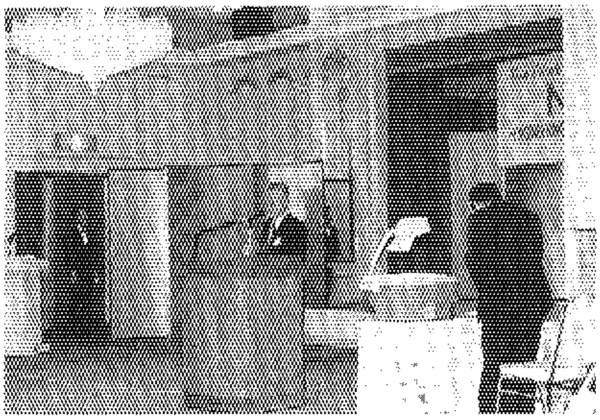
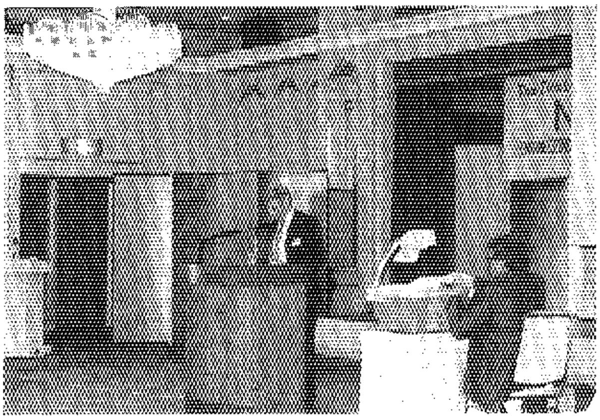
International Symposium
NUCEF '95

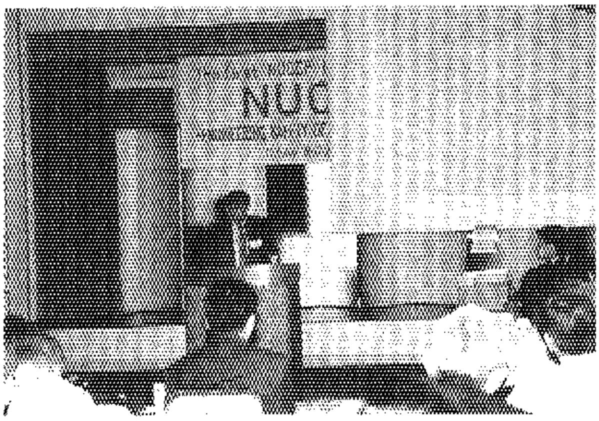



International Symposium





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