TOPICAL MEETING

Safety of the Nuclear Fuel Cycle

Cadarache, France – 20-21 September 1994
SAFETY OF THE NUCLEAR FUEL CYCLE

Topical Meeting
Cadarrache, France (20-21 September 1994)

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT
Paris 1996

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The primary objective of NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

— encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;
— assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;
— developing exchanges of scientific and technical information particularly through participation in common services;
— setting up international research and development programmes and joint undertakings.

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also cooperates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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On behalf of the Director of the Institute for Nuclear Safety and Protection (French abbreviation IPSN), I welcome all the participants in this meeting. The IPSN is an institute which is dependent on the Atomic Energy Commission but which is greatly autonomous in comparison with the other sectors of this organization. It performs research and development tasks (such as the Phebus Fission Products programme conducted at the nearby Cadarache research centre) and expert appraisals in particular on behalf of the government authorities and more especially the safety authority, namely the Nuclear Installations Safety Directorate.

Talking about nuclear safety is first and foremost associated with the risks represented by nuclear reactors and immediately calls the Three Miles Island and Chernobyl accidents to mind. Besides, specialists in nuclear reactor safety have the opportunity to meet and deal with their problems during frequent international conferences.

On the contrary, the safety of fuel cycle installations arouses less interest, except for waste storage but this subject is not part of this meeting. The accidents which occurred in fuel cycle facilities did not cause a great stir in the media and the safety of such installations is rarely discussed in international meetings. Nevertheless, criticality experts meet regularly.

However, risks are not missing as shown by a number of accidents which had or might have had serious consequences on the environment, the public or workers. Without trying to draw up an exhaustive list, we can quote

- the Kyshtym accident in Russia which led to the dispersion of a large amount of fission products in the environment,

- the COMURHEX accident in France which led in 1977 to the release of 7 tons of uranium hexafluoride into the environment as people on holiday were driving to the south of France on the nearby motorway,

- the Kerr Mc Gee accident in the USA which was due to the explosion of an uranium hexafluoride container and killed a worker,
- and more recently - I think it will be discussed in a little while- the accident which occurred at Tomsk in Russia in 1993.

Of course such accidents, as already said, did not create a long and considerable stir as in the case of some reactor accidents but is should all the same be borne in mind that in fuel cycle installations the amounts of radioactive materials used or stored are sizeable and much higher than those contained in a nuclear reactor if I consider, for example, the case of reprocessing plants. Only the processes implemented allow possibilities of dispersion lower than for nuclear reactors. Nevertheless, explosion risks - and this gives you food for thought - should deserve greater attention than previously since most of the significant accidents are of an explosive type. Furthermore, considering the operating characteristics of fuel cycle installations, particular attention must be paid to the consequences of accident situations for workers who are roughly, except for the above-mentioned accidents, the only people really exposed in case of accident; and these consequences may obviously be particularly tragic.

It seems to me therefore that the initiative taken by the OECD working group on fuel cycle safety for this meeting, through the impetus given by its chairman, Dr BROWN, is particularly fortunate. This meeting follows a few other ones which were held at Salamanque in Spain in 1986 and in Tokyo in 1991. I think that this type of meeting makes it possible to extend and to go deeper into the working group's action. It must enable the specialists from the various countries to share their experience and be a forum for fruitful exchanges. At the overall stage of maturity reached by the nuclear industry, I think that the use of operating experience must be a driving force for practical improvement of the safety of installations. I invite you thus to in-depth exchanges on experience gained by each other through both the papers presented and of course the exchanges which will be made subsequently outside the meeting. Thank you.
Opening remarks by Dr M L Brown

I feel personally honoured and pleased to welcome delegates to the meeting on behalf of the fuel cycle safety working group of the CSNI. A particular welcome to our Russian colleagues whose attendance has been facilitated by the IAEA as co-sponsors of the meeting. I would also like to thank IPSN, and Monsieur Auchere in particular, for their help and hospitality in organising the meeting here.

As Monsieur Queniart notes in his remarks, the fuel cycle processes rarely give rise to accidents of significant consequences off site. Nonetheless handling mobile radioactive materials is an intrinsically hazardous activity and it is necessary to strive to apply best technology to keep the already good standards improving. This was indeed the main conclusion of the recent state of the art report on safety of the nuclear fuel cycle published by the OECD.

That Report did identify areas of developing technology, however, and the working group took the view that exchange of experience on operating plants and research plans for the future were both important contributions to safety. This seminar continues from earlier meetings the information exchange; the topics covered include plant experience, safety assessment and techniques, and safety research. I look forward to a successful meeting and fruitful interchanges.
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
WORKING GROUP ON FUEL CYCLE SAFETY
TOPICAL MEETING
SAFETY OF THE NUCLEAR FUEL CYCLE

CADARACHE, FRANCE
20 - 21 SEPTEMBER 1994

OECD/NEA NUCLEAR SAFETY DIVISION

OECD/NEA NUCLEAR SAFETY DIVISION

WHAT IS THE OECD?
WHAT IS THE NEA?
COMMITTEES WITHIN NEA
SAFETY COMMITTEES WITHIN NEA
WORKING GROUP ON FUEL CYCLE SAFETY

OECD/NEA NUCLEAR SAFETY DIVISION
B. Kaufer September '94
WHAT IS THE OECD

- ORGANISATION FOR ECONOMIC AND COOPERATIVE DEVELOPMENT

- AN INTERNATIONAL ORGANISATION OF 25 COUNTRIES

- A TOOL FOR INTERGOVERNMENTAL COOPERATION, MAINLY IN THE ECONOMIC FIELD

- PLACE FOR POLICY MAKERS TO COMPARE POINTS OF VIEWS AND EXPERIENCES

OEC D COUNTRIES

- Australia
- Austria
- Belgium
- Canada
- Denmark
- Finland
- France
- Germany
- Greece
- Iceland
- Ireland
- Italy
- Japan
- Luxembourg
- Mexico
- Netherlands
- New Zealand
- Norway
- Portugal
- Spain
- Sweden
- Switzerland
- Turkey
- United Kingdom
- United States
AIMS OF THE OECD

- PROMOTE POLICIES DESIGNED:
  - TO ACHIEVE HIGHEST SUSTAINABLE ECONOMIC GROWTH
  - TO CONTRIBUTE TO ECONOMIC EXPANSION
  - TO CONTRIBUTE TO EXPANSION OF WORLD TRADE

WHAT IS THE NEA

- NUCLEAR ENERGY AGENCY
  - ONE OF 15 BODIES OF THE OECD
  - IT'S AIM IS TO PROMOTE THE DEVELOPMENT OF NUCLEAR ENERGY AS A SAFE, ENVIRONMENTALLY ACCEPTABLE ENERGY SOURCE
  - TECHNICAL COMMITTEES
TECHNICAL COMMITTEES

▷ RADIOACTIVE WASTE MANAGEMENT COMMITTEE (RWMC)

▷ COMMITTEE ON RADIATION PROTECTION AND PUBLIC HEALTH (CRPPH)

▷ COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS (CSNI)

▷ COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES (CNRA)

TECHNICAL COMMITTEES

▷ GROUP OF GOVERNMENTAL EXPERTS ON THIRD PARTY LIABILITY IN THE FIELD OF NUCLEAR ENERGY

▷ COMMITTEE FOR TECHNICAL & ECONOMIC STUDIES ON NUCLEAR ENERGY DEVELOPMENT AND THE FUEL CYCLE (NDC)

▷ NUCLEAR SCIENCE COMMITTEE (NSC)
SAFETY & LICENSING COMMITTEES

- COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS (CSNI)
  - Senior Scientists and Engineers

- COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES (CNRA)
  - Senior Regulators
CSNI RESPONSIBILITIES

- TECHNICAL ASPECTS OF DESIGN, CONSTRUCTION AND OPERATION
- REVIEW OF SELECTED SAFETY ISSUES
- PROGRAMMES TO RESOLVE DISCREPANCIES
- ESTABLISHMENT OF JOINT PROJECTS

OECDB/NEA NUCLEAR SAFETY DIVISION
B. Kaufer September '94

SESAM
Senior Group of Experts - Severe Accident Management

- Created in 1989 to study the potential response of LWR systems to severe accidents.
- Through the issuance of several reports has provided a consensus technical view regarding Member countries' SA research policies, positions regarding SA, and SA management programme development.

OECDB/NEA NUCLEAR SAFETY DIVISION
B. Kaufer September '94
To guide the operation of the Incident Reporting System, identify significant technical safety issues, and supervise specific activities in the human factors area, including operator training.

Recent Activities and Achievements

- IRS - Current NEA-IRS data base contains some 2150 reports.
- Generic Studies - Studies on Incidents related to reactivity, loss of RHR, and loss of containment functions.

Human Factor Reports:
- Improving Human Performance on maintenance activities
- Man-machine interface
- Analysis of incidents involving cognitive errors and erroneous human actions

Workshops:
- Fire protection and fire protection systems
- Motor Operated Valves
PWG 2
Coolant System Behavior

- Oversee studies of phenomena within the reactor coolant system boundary for the whole spectrum of core accidents

- Task Groups:
  - Thermal-Hydraulic System Behaviour
  - In-Vessel Degraded Core Behaviour
  - Small ad-hoc groups for specific tasks (i.e., Accident Management Writing Group, Code Validation Writing Group, etc.)

OECD/NEA NUCLEAR SAFETY DIVISION
B. Kaufer September '94

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PWG 3
Reactor Component Reliability

- Topics related to the safety and structural integrity assessment of reactor components, particularly of the primary circuit components.

- Fracture Assessment Sub Group which has organised FALSIRE round robin and Workshops in the area of analysis.

- Three main areas considered:
  - Inspection (recently completed OECD PISC project)
  - Analysis (FALSIRE)
  - Materials Properties (metallurgy and aging issues)

OECD/NEA NUCLEAR SAFETY DIVISION
B. Kaufer September '94
- Exchange information in the area of severe nuclear reactor accident phenomena occurring predominantly outside the reactor vessel

- Exchange of detailed information and discussions (groups of experts & specialist meetings) of progress achieved in respect to specific technical issues

- Review and summarise results, assess level of knowledge in regard to requirements of nuclear safety, and develop technical positions

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- Three major Task Groups
  - Fission Product Phenomena in the Containment
  - Severe Accident Phenomena in the Containment
  - Aspects of Severe Accident Management

- Since 1981, PWG 4 has organised five International Standard Problems covering core/concrete interaction, hydrogen distribution, and fission product transport in containment

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B. Kaufer September '04
PWG 5
Risk Assessment

- Deals with the technology and methods for identifying contributors to risk and assessing their importance
- Exchange information on current research in the area of probabilistic risk assessment
- Recent reports include:
  - State of the Art of Level 1 PSA Methodology
  - PSA Application to Technical Specifications
  - Shutdown and Low Power Safety Assessment
  - PSA of LWR Containment Systems Performance

OSGD/NIA NUCLEAR SAFETY DIVISION

PWG 5
Risk Assessment

- Current Task Groups:
  - Quantitative Safety Guidelines
  - Status Of PSA Programmes
  - Fundamental Principles of Living PSA
  - Plant Specific Data Collection
  - Non Full Power Operation & Shutdown Conditions
  - Risk Based Management of Safety Reliability
  - Human Interactions: Critical Operator Actions & Data Issues
  - Level 2 PSA Methodology and Severe Accident Management
  - PSA Based Plant Modifications and Backfits

OSGD/NIA NUCLEAR SAFETY DIVISION

B. Kaufer September '94
CNRA RESPONSIBILITIES

- IMPACT OF NEW SAFETY TECHNOLOGY INFORMATION ON REGULATION

- EXCHANGE VIEWS ON EVOLUTION OF REGULATIONS

- SHARE RATIONALE FOR ANTICIPATED REGULATORY MEASURES

OECD/NEA NUCLEAR SAFETY DIVISION

B. Kaufer September '94

WORKING GROUP ON INSPECTION PRACTICES

- Focus of WGIP is on safety inspections carried out by on behalf of the regulatory authorities.

- Main purpose is to facilitate the exchange of information and experience on safety inspections.

OECD/NEA NUCLEAR SAFETY DIVISION

B. Kaufer September '94
WORKING GROUP ON
INSPECTION PRACTICES

- Activities
  - Report being prepared on Inspection Training and Practices
  - Report being prepared on Framework of Inspection Philosophy, Organisation and Practices
  - International Workshop held in Helsinki, Finland - May 1994.

WORKING GROUP ON
FUEL CYCLE SAFETY

- MANDATE:
  - To prepare a state-of-the-art report on the safety of the fuel cycle identifying critical safety issues and indicating where further work is required.
  - To periodically review the situation in the field of fuel cycle safety and recommend priority actions.
WORKING GROUP ON
FUEL CYCLE SAFETY

ACTIVITIES:

- Publication - The Safety of the Nuclear Fuel Cycle - 1993

- Fuel incident Notification and Analysis System (FINAS) - Started in 1992 - Data Base contains 24 reports.

- Meetings and Symposium
  - Belgium 1992
  - Cadarache 1994
  - Japan 1995
TOPICAL MEETING ON THE SAFETY OF THE NUCLEAR FUEL CYCLE

N. OI
International Atomic Energy Agency
Head, Nuclear Materials and Fuel Cycle Technology Section

Thank you Mr. Chairman, ladies and gentlemen!

In the name of the International Atomic Energy Agency, as a cooperating organization, it is a great pleasure for me to welcome all of you to the Topical Meeting on the Safety of the Nuclear Fuel Cycle organized by NEA/OECD.


You may be aware of the fact that there is a lot of criticism from Member States concerning the duplication of activities among international organizations. It is even more appealing when there is a serious financial squeeze in international organizations. Since the IAEA has also been working for many years on various aspects of the safety of nuclear fuel cycle, we are here to communicate and coordinate the activities in order to avoid possible duplication.

On the occasion of this interesting Topical Meeting, we are very pleased that two prominent Russian experts are here as an extended delegation of the IAEA to present papers on the TOMSK-7 accident and the safety and performance of the Russian reprocessing plant RT-1.

I wish you all a pleasant, successful and fruitful meeting.

Thank you!
TABLE OF CONTENTS / SOMMAIRE

SESSION 1 - OPERATIONAL SAFETY IN NUCLEAR FUEL FACILITIES

1 SAFETY PERFORMANCE OF THE FAST REACTOR REPROCESSING PLANT AND SUPPORT FACILITIES 1988 - 1994
   MR. T. BARRETT, MR. K. SINCLAIR, MR. D. STEWART, MR. P. PAGE, MR. A. ANDERSON AND MR D. SINCLAIR

2 OPERATIONAL EXPERIENCE AT THE TOKAI REPROCESSING PLANT
   MR. H. IWABUCHI, MR. T. NAKAJ AND MR. M. KANAMORI

3 TOMSK - NUCLEAR EVENT - CAUSES, CONSEQUENCES AND LESSONS LEARNED
   DR. E.G. KUDRIAVTSEV

4 MANAGEMENT OF EXTERNAL DOSES AND RISK OF CONTAMINATION IN BELGONUCLEAIRE PLANT FOR FABRICATION OF MOX FUEL AT DESSEL
   MR. P. KOCKEROLS AND MR. D. VAN DEYCK

5 A PROGRAMME FOR THE ASSESSMENT OF RADIOLOGICAL RISK ASSOCIATED WITH ACTIVE HANDLING OPERATIONS
   MR. D. L. HAMBLEY

SESSION 2 - SAFETY CRITERIA AND REGULATORY PHILOSOPHY

1 SAFETY REQUIREMENTS FOR NUCLEAR FUEL CYCLE FACILITIES IN GERMANY
   DR. B. GMAI, MR. W. HEINICKE AND MR W. THOMAS

2 REVISION OF US DOMESTIC LICENSING OF SPECIAL NUCLEAR MATERIAL, 10 CFR PART 70
   DR. M. KLASKY, MR. T. COX AND DR. R. MILSTEIN

3 LICENSING OF THE SELLAFIELD MOX PLANT SMP
   DR. R.J. PAGE, MR. A.J. TILSTONE

4 SAFETY REGULATIONS FOR NUCLEAR FUEL FACILITIES IN JAPAN
   MR. K. HIROSE
SESSION 3 - PLANT HAZARD ANALYSIS AND MITIGATION

1 TECHNOLOGICAL ASPECTS OF SAFE OPERATION OF REPROCESSING PLANT RT-1 IN RUSSIA

2 DESIGN MEASURES TAKEN AGAINST THE RISK OF SPREADING RADIOACTIVE MATERIALS IN FUEL CYCLE FACILITIES IN FRANCE

3 SAFETY ASSESSMENT OF FIRE POSTULATED IN A PROCESS CELL OF A FUEL REPROCESSING PLANT

4 MODERATELY DENSE CONCENTRATION - CRITICALITY SAFETY

5 STUDY ON SOME PROBLEMS FOR CRITICALITY SAFETY EVALUATION IN JAPAN

6 SAFETY RESEARCH PROGRAMS IN NUCEF FOR FUEL CYCLE BACK-END FACILITIES
SESSION 4 - PLANT EXPERIENCE AND EMERGENCY PLANNING

1. OPERATIONAL REVIEW OF THE FAST REACTOR REPROCESSING PLANT 1979 - 1994

2. DESIGN BASIS OF OFF-SITE EMERGENCY RESPONSE PLANS FOR FUEL CYCLE INSTALLATIONS

3. IMPACT OF THE LA HAGUE REPROCESSING PLANTS ON THE SURROUNDING ENVIRONMENT

4. ACTIVITIES ON PUBLIC RELATIONS WITH RESPECT TO FUEL CYCLE SAFETY IN JAPAN

MR. T. BARRETT, MR. D. STEWART, MR. A. ANDERSON AND MR. D. GORDON

MR. J.P. RZEPKA, MR. P. DUBAUX, MRS. A-C. JOUVE, MR. T. CHARLES AND MR. J-P. MERCIER

MR. J. SIMONNET

MR. K. HIROSE
SAFETY OF THE NUCLEAR FUEL CYCLE - TOPICAL MEETING

OECD NUCLEAR ENERGY AGENCY, CADARACHE, SEPTEMBER 1994

SAFETY PERFORMANCE OF THE FAST REACTOR REPROCESSING PLANT
AND SUPPORT FACILITIES 1988 - 1994

by

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SUMMARY

The Fast Reactor (FR) 'raison d'etre, and the reprocessing and related facilities are briefly described together with the changes in operating philosophy and objectives over the five years. The safety performance of a wide spectrum of the facilities - addressing both conventional and radiological hazards - is presented and discussed. The event and incident reporting arrangements are discussed, and some are outlined where they led to significant changes in practice.

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(3) UKAEA, Technical Assessment and Safety Department, NSO, Dounreay
Dounreay, Thurso, Caithness, Scotland, United Kingdom
INTRODUCTION

FR Project Facilities at Dounreay

1. The FR reprocessing and related facilities at Dounreay, and operational aspects, have been described extensively in the open literature (1, 2, 3, 4, 5 and 6) so details are not repeated here.

2. In summary the major PFR fuel cycle facilities at Dounreay include:-

(i) the Fast Reactor Reprocessing Plant - with its associated product and export facility (FRRP) and latterly its extension to include a dedicated fuel residue recovery unit (RRP); (ii) the Highly Active Liquor Storage and evaporation plant (HALS) with an associated MA liquor treatment plant (FLOC); (iii) the Low Active Liquor Drains (LAD) and effluent tankage and discharge system (LA Tanks); (iv) the High ‘alpha/beta gamma’ solid waste - intermediate level (ILW) store - and the later immobilisation cementation plant for overpacking waste drums (DCP); (v) the High ‘alpha/low beta-gamma’ waste (PCM) store; (vi) the low ‘alpha’ high beta gamma waste receipt facility (Silo); (vii) the low active solid waste disposal pits (LAWP); (viii) the uranium recovery plant; (ix) the highly active laboratory (PAL); (x) the low active laboratories (CSG Labs).

3. Many of the facilities for the PFR reprocessing project were generated by the reconstruction of former DFR fuel cycle related plants and laboratories. This involved extensive decommissioning work and led to the gaining of very valuable early experience in the field, much of which has been reported (1, 7). That also led to the formulation of chemical plant decommissioning design philosophies (8) many of which were then accommodated within the plants being built - or rebuilt - for the PFR project.

4. Of the facilities identified earlier only two were totally new ‘green-field’ design and construction exercises. Those were the - High alpha/Low beta-gamma PCM waste store, the Low alpha/High beta-gamma waste Silo, and the cementation plant.

Objectives and Strategy

5. The early FR project objectives and strategy were wide ranging and were primarily:-

a) The demonstration of a full commercial Fast Reactor operating fuel cycle - necessary for the utilities.

b) The minimisation of the PFR plutonium fuel inventory.

c) The establishment of a process efficiency database.

d) The demonstration of the application of safeguards - the nuclear material accountancy aspects.

e) The exploration of the technology leading to further development work for potential commercial application.

f) The gaining of safety and environmental related experience.

6. The strategy was further developed after the commencement of the reprocessing operations in 1980 to include international collaboration.
7. The objectives have been modified over the fourteen years of operating the cycle, with a shift of emphasis to achieving high levels of performance - particularly in relation to safety, safeguards and environmental issues. Further emphasis has been given to plant/process efficiency - the desire to operate at an optimum load factor - to realise financial economies.

8. A very significant level of effort has been expended since 1988 in relation to the more formal regulatory licensing placed on the UKAEA in 1990. Prior to that date the UKAEA had been, as a government department, exempt from regulatory licensing requirements - albeit they were required to operate to the same standards. The introduction of the licensing regime - from October 1990 - led to a major review and reassessment of facility risk assessments and the establishment of revised safety management arrangements. The quantified risk assessments have demonstrated that the facilities are well within the UKAEA specified requirements and the UK levels identified by the Nuclear Installations Inspectorate - which in turn conform to EEC and International standards.

2. OPERATING HISTORY

9. Since 1980 the Dounreay FR reprocessing plant has reprocessed over 23 tonnes of Fast Reactor fuel and extracted over 3½ tonnes of plutonium. This has been obtained from 239 sub-assemblies, 144 mixer breeders and 10 radial breeders and has included 4.2 te of unirradiated fuel and fabrication residues. The highest fuel burn-up processed to date was 17.6%; core assemblies are routinely processed at 10-12% burn-up.

10. Routine reprocessing has caused no intractable problems. During the first fourteen years the fuel reprocessing plant kept pace with reprocessing irradiated fuel as it arose from the reactor. The throughput has generally increased progressively in line with the steady improvements in the PFR operating performance and the increasing station load factor, and with increasing levels of residue recovery. This is illustrated in Table 1.

11. The operating history of the main FR reprocessing plant is not discussed further as it has been identified before (Ref 6) and is discussed in the companion paper being delivered to this conference.

12. The related fuel cycle plants operated mainly to meet the requirements of that main reprocessing programme, generally in line with the throughput illustrated in Table 1. In some cases the workload in the associated facilities is required to parallel the main plant (eg analytical laboratories) in others some delayed phasing is acceptable (eg Medium Active Liquor Floc).

3. RADIOLOGICAL AND ENVIRONMENTAL PERFORMANCE 1988 - 1994

13. The plants and related facilities had progressively improved their radiological performance when related to throughput over the first two decades of the site operations - from 1958 to 1978. This is illustrated for example by the liquid effluent alpha activity discharges - Figure 1; this topic has been discussed already (Reference 6).
14. The first major improvements were initiated in the late 1960s as part of a programme to ensure that individual dose exposures would not exceed 5R (50 mSv) per year. These improvements were concentrated on the reprocessing plant but were adopted for all facilities who took advantage of technological `spin-off’. Associated with this initiative, the safety performance of the facilities was then kept under additional regimens of surveillance. A second initiative was associated with the adoption of the PFR reprocessing programme with the consequential decommissioning, reconstruction and new build activities.

15. The approach to the safety design of the ‘new’ facilities was essentially two fold:- to ensure that the additional feature required for technical reasons were meeting modern - and projected - safety standards, and to accommodate design features that the ‘5R’ studies had indicated required attention but where access constraints had precluded them previously being tackled.

16. The plant/process improvements did lead to improved safety performance. This was noticeable immediately the facilities were brought back on line in 1980 in the PFR reprocessing related mode. Continuing surveillance and regular review saw steady improvements year by year up to the late 1980s; particularly when related to plant throughput levels and inventories.

17. It is stressed that since the mid 1980s it has been attention to detail that has been necessary to secure further improvements, together with major operational and hardware performance reviews leading to major engineering modifications (eg changes to Master Slave Manipulator (MSM) management arrangements, the cage crane improvements).

18. The levels of safety performance achieved over the last six years are illustrated in a series of figures, these should be read in conjunction with the Figure 2 graphically representing the load factor/throughput since 1980.

19. Figure 3 illustrates the operator dose uptake over the years since 1988. The records over these years indicate a generally stable situation. They confirm that dose uptake is less related to plant load factor than it is to personnel occupancy within the plants whilst carrying out engineering developments. This has placed increasing emphasis on dose budgeting in relation to projected plant improvements to ensure that future benefits are more than commensurate with near term detriments; this also addresses the issue of maintaining personnel exposure as being ‘ALARP - As Low As Reasonably Practicable’.

20. Figure 4 illustrates the activity in the liquid discharges from the site. Apart from minor perturbations arising from special activities, these tend to be dominated by the reprocessing and related activities. Comparison to Figure 1 confirms the most satisfactory improvement since the mid 1970s consequent upon the many improvements introduced since then.

21. Figure 5 illustrates activity discharges to atmosphere. These are in line with the observations made above.
4. PERFORMANCE & DIFFICULTIES

22. As identified earlier the general 1970s design intent has been achieved in practice. The achievement of further improvements has required attention to detailed features - operational, engineering and safety/environmental.

23. Much of this attention has been by routinely assessing performance in relation to the original design safety documentation and later developments in standards; a particular focus was the production of new Safety Cases in 1988-1990 where total safety reviews were re-enacted ‘from square one’. Close formal surveillance has continued since 1990 with an additional need to satisfy regulatory bodies on the performance and on many development proposals.

24. A further input to improving performance is the attention given to investigating unexpected events; this is discussed later. In addition, performance difficulties are kept under continued surveillance and are responded to.

Fuel Disassembly

25. The ‘as-built’ cave facility has been described (Reference 9) and some operational experience has been discussed (References 2, 5 and 6). Some of the more significant developments are discussed below.

26. Sparks arising from the laser cutting operations have the potential to ignite any combustible material which may be nearby. Any fire in the cell has the potential to increase the cell ventilation HEPA filtration loading, leading to increased filter changing frequency. The avoidance on the use of combustible material and the provision of a remotely operated BCF fire extinguishers have ensured that this is not a problem.

27. In common with other in-cave cranes, the fuel disassembly crane has required a considerable amount of maintenance to repair failures resulting from radiation damage to cables, plugs, sockets and junction boxes. The addition of shielding, and some remote handling gear to the crane maintenance glovebox combined with remotely operated plugs and sockets on the crane have been major features in the dose reduction strategy. The drainage of the glovebox allowing water jetting of the crane to be utilised as part of this strategy, has been demonstrated to be most effective. A major improvement was the replacement of the crane by a largely stainless steel constructed unit.

28. The fuel was originally cropped by a hydraulically operated unit with the power unit located outside the shielding. The cropper suffered from a series of failures of solenoids, valves and seals in its early years. The seal failure allowed the hydraulic oil to become contaminated from the air inside the cell, resulting in an increase in the dose rate to the operators at the cave window. The hydraulic cropper was satisfactorily replaced with an in-cave powered cropper.
29. Use of Master Slave Manipulators (MSMs) generally results in three principal problem areas:

(a) mechanical failures
(b) gaiter failures, and
(c) decontamination problems.

and it was in recognition of this that the design intent was to minimise demands on these units. Overloading of the manipulator is the principal cause of failures and more robust manipulators have been assessed, and some installed.

30. Operational experience has emphasised the need to remove MSMs before extensive gaiter failure in order to minimise operator dose consequent upon the subsequent decontamination needs.

Ventilation Systems

31. Experience had demonstrated that attempts fully to empty high active liquor containing vessels using either fluidic pumps or steam ejectors may cause activity to be levitated into the plant ventilation system. This may be prevented by leaving a heel of liquor in the bottom of any vessel being transferred to another vessel and additional design controls were installed to achieve this.

32. The balance of ventilation systems is an area which requires continued vigilance. In laboratories containing separate fumehood and glovebox extracts it is important to design the systems such that failure of one system does not result in a reverse flow being set up by the remaining operating system. In plant systems with interlinked vessel and cell ventilations - it is important to maintain the correct balance - care is required to ensure that temporary extract systems do no cause a reverse flow allowing active air...to be pulled into the working environment. Drain ducts leading outside buildings can, under certain adverse weather conditions, permit reverse airflows into (for instance) active areas, unless sufficient precautions have been adopted at the planning stage.

Floc Process Failures

33. The bulk of the early uranic discharges from the reprocessing plants were subject to a precipitation process whereby the ammonium di-uranate flocc is employed to decontaminate other active effluent streams. The presence of complexing agents in the medium active liquor prevents the flocculation of activity due to the stronger chemical bond between the radioactivity and the complexing agent. No liquor containing complexing agents is permitted in this process stream and control arrangements have been installed to prevent the introduction of those, and of detergents.

34. The floc resulting from this process is stored in stainless steel clad concrete tanks. One of these tanks has developed a failure such that minute quantities of activity can be detected on the concrete surface of the tank. The lack of secondary containment at the design stage led to this problem; regular monitoring up ensures that this is not a safety problem. Further tankage is being installed to extend the capacity.
Laboratories

35. The laboratories have operated well, providing support on demand to the reprocessing and waste plants.

36. Problems have been experienced with the fumehood ventilation ducts being corroded by the acidic vapours generated during some of the analytical work. Corrosion resistant ducting was installed.

37. Glovebox problems associated with achieving adequate demonstration of leak-tightness have proved to be inconvenient. This is caused by the multitude of penetrations into the box. Older mild steel gloveboxes have suffered from corrosion after long periods of use with an acidic environment. Care is taken to ensure that potentially corrosive acids are adequately treated prior to discharge to the stainless steel drain network to minimise drain corrosion.

38. Whilst the use of shielded and unshielded La Calhene containers has successfully minimised the generation of secondary waste, experience has demonstrated that their use requires careful management. Contaminated seals are a problem - container or port lugs require careful management to ensure satisfactory system performance.

39. The practice of returning excess samples to the plant of origin has resulted in a significant reduction in activity discharged to the medium active liquor stream. This has resulted in a consequential reduction in the activity discharged from the site.

Low Active Waste Disposal Pits

40. The successful utilisation of available space has always been a key feature of operating the Low Active Waste Pits. The introduction of the Supercompactor in 1990 was a significant milestone in the operation of this facility. A volume reduction of a factor of 5 is frequently achieved, and this has considerably improved the space utilisation in these pits. To further improve volume utilisation an alligator cropper was purchased to size reduce bulky waste items.

41. Rainwater ingress into the in-use pit has been reduced by the provision of a cover building, and this has reduced the volume of groundwater which has to be treated by the site effluent system. Waterproof membranes were earlier installed on the top of the closed pits - tests demonstrated that there was no methane build-up problems.

5. SAFETY RELATED DIFFICULTIES

42. The Fast Reactor Reprocessing Plant has operated from 1988 with few major safety related issues; this justifies the original safety assessments and design philosophy. The recorded minor safety related incidents have, however, increased partly due to an increasing operational usage of the plant, but mainly due to an increased awareness of the value of the event reporting system. The minor events and the value derived from reporting and investigating them are discussed in this section.
Event and Incident Reporting

43. The incident reporting system used in the early period covered by this paper (1988-1990) only categorised "Incidents" as reportable centrally. Minor plant occurrences were reported locally as "Events". These last included a very wide range of minor happenings which were investigated by the building management, and if relevant the local Safety Coordinator.

44. As identified previously in this text, one of the principle ways forward on safety and related improvements is to attend to detail and this need is focused on by a monitoring, reporting and reviewing regime. It is a requirement that a facility manager has a documented 'Authority to Operate' (ATO) which requires annual renewal. To achieve that renewal the ATO Holder is required to demonstrate that an effective safety management arrangement is in place, to review the performance, and to identify the forward programme.

45. Part of this wide ranging review includes a study of incidents and events (the near misses), that is additional to the radiological and environmental considerations driven by the ALARA (or ALARP in the UK) philosophy.

46. In 1991 a unified site UNOR (Unusual Occurrence Report) system was adopted. The new system categorised incidents from Grade 1 (very minor) to Grade 5. Incidents raised as Grade 3 and above are reportable to the UK regulatory bodies within a mandatory time/period. The new Grade 1 UNOR - in effect the 'near misses' replaced the old "Event" and brought the reporting of minor occurrences into a unified, focused procedure. The trends of the reporting of incidents can be seen in the Figures.

47. Every encouragement is given to staff to report Grade 1 UNOR events. The reporting does reveal root causes that under other circumstances could be less favourable - identifying them does enable early attention to potential real problems. This advantage has been demonstrated and reporting attention is continuing to improve. The system is of greatest benefit if the events are investigated and placed on record and are disseminated widely beyond the immediate plant.

Overall Frequency of Events

48. In 1992 and 1993, the PFR reprocessing group of facilities generated 143 and 181 UNOR reports retrospectively. These were 25-30% of the 500-550 UNORs recorded within the Fuel Cycle Area (FCA). The records indicate that the UNORs were spread evenly across the reprocessing plant, the laboratories and the remaining support facilities - each at one third.

49. The UNOR system has identified the contribution to events of human and/or engineering shortcomings. The distribution of them across the group range from - the laboratories where the nature of the work is very much hands-on and the incidence is largely handling and human error related - to the main plant where in addition to human error the engineering difficulties predominate. The majority of the UNORs arise in the high workload facilities, whether that be a throughput high load factor or extensive engineering attention.
50. Generally speaking events and incidents are confined to human error and/or equipment failure. The former can include an element of confusion and over willingness; equipment failure can include a human component in the optimum response not being evident - or followed by - the operator.

Overall Trends

51. The charts show that while the frequency of Category 2 (and greater) incidents within these plants has remained generally constant throughout the review period, there has been an increase in reported Category 1 events, particularly since 1991. This trend reflects the introduction and implementation of the expanded (in scope) reporting system and an increased awareness of the need to report events. This is the basis of the means of identifying root causes of accidents and mechanisms to avoid them. The trend has continued, certainly over 1990 to 1993, but the projection for 1994 would indicate a modest decrease from the 1993 figures.

Injuries and other Industrial Events (1988 - 1993)

52. These are recorded in the UNOR system as an identified class of incidents (g). Certain injuries/events are required to be reported to the UK Health and Safety Executive (HSE); reportable injuries are included at level 2 or above in the UKAEA UNOR system. During the period 1988-1993 there were no reportable injuries in the area.

53. Assuming that 1 'man year' represents 46 weeks at 37.5 hours per week (i.e. 1725 hours per year in the industrial environment) the figures equate to the following accidents per $10^6$ man hours worked.

<table>
<thead>
<tr>
<th></th>
<th>Reprocessing Plant</th>
<th>Laboratories</th>
<th>Other Facilities</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>G1</td>
<td>G2</td>
<td>G1</td>
</tr>
<tr>
<td>1988</td>
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<td>7.7</td>
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<tr>
<td>1989</td>
<td>0.0</td>
<td>7.6</td>
<td>0.0</td>
</tr>
<tr>
<td>1990</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>1991</td>
<td>8.5</td>
<td>8.5</td>
<td>0.0</td>
</tr>
<tr>
<td>1992</td>
<td>44.6</td>
<td>0.0</td>
<td>33.3</td>
</tr>
<tr>
<td>1993</td>
<td>36.8</td>
<td>0.0</td>
<td>49.7</td>
</tr>
</tbody>
</table>

Industrial Events, Other than Those with an Inquiry Potential

54. These accidents are typically equipment which has ceased to function or has broken down through component failure. The group of events include those where the malfunction has been caused by human errors.
55. The value is such recording and analysis is only of real consequence if it leads to remedial action. As well as studying generalities and trends, the arrangement also includes a detailed investigation into the event and then that being reported widely across all plant management.

56. The general and trend review of itself has produced tangible benefits. There were early indications of an excess of manual handling events; many of these were in line with HSE observations - with subsequent regulation enactment - and have been countered by the provision of extra lifting and handling facilities, others were identified as leading to minor hand injuries - where the increased use of protective gear has now been seen to check this trend.

57. The individual event investigations meet the needs whereby performance improvements are now more related to detail considerations. The follow up and information dissemination avoid similar, but possible worse, situations arising again and in other facilities. This is exemplified by the attention given recently to extraneous liquor arisings and liquor mis routings.

58. Events incurring substantial financial cost, including plant outages, mirror these involving accidents and are at about 66% of the frequency of the industrial injuries. Minor 'system' malfunctions are more prevalent, (between 3 and 4 times as frequent as cost-incurring events) and the pattern of these follows that of the previous 2 groups.

6. MANAGEMENT OF SAFETY ARRANGEMENTS

59. Developments within the fuel cycle activities are pursued and prioritised on the basis of several interactive issues; broadly these encompass:- safety performance, environmental aspects, commercial considerations and process efficiency, regulatory requirements, and reviews against modern standards developed for the relevant technology.

Safety & Environmental

60. Safety and environmental issues are addressed in similar ways so are discussed here together. It is noted that safety is not confined to radiological aspects, the 'industrial type' safety is given a high profile as it is in this respect that most real harm is occasioned. Furthermore, safety issues are addressed primarily in their own right to ensure that safety performance is not allowed to suffer; the legislative requirements are seen as being a corollary to that.

61. The managers and operatives in each facility are required to keep safety and environmental matters to the forefront of their considerations and are required to keep performance under continued surveillance. The managers additionally are required to operate their facilities within well defined bounds, set out clearly in a suite of documents enshrined within a formally issued 'Authority to Operate' (ATO).
62. The review function features as part of a total safety management system that is based on a UKAEA corporate arrangement and which addresses all aspects relevant to safety, environmental and regulatory requirements. The requirements are promulgated within a Quality Assurance (QA) system by means of Corporate Instructions, Local Instructions, Local Procedures and then plant specific Directives, Standing Instructions and Plant Operating Instructions (POIs).

63. Great emphasis is placed on training and ‘refresher’ retraining much of which is based on the requirements enshrined within the above safety management system. In particular the operative training is effected in small part by a classroom approach to general principles and then ‘on-the-job’ detailed process aspects based on the detailed POIs; these together generate a training profile for every member of staff. All training is recorded on a widely accessible computer database; this has in part been driven by the regulatory needs but has been recognised as having very effective personnel deployment advantages. The record is related to the profile for each person and the review of that is a major part of the commitment to training.

64. All nuclear plants have comprehensive Safety Case reports addressing all aspects of nuclear and radiological safety; these are now being extended to cover all of the more common ‘industrial’ safety issues. These were required to meet regulatory needs and have been a most expensive exercise in their generation and continued upkeep. Their preparation - in 1988 to 1990 - did require a total safety review for each plant which did reveal a few imperfections which were then addressed. Whether this was the most cost effective way of effecting these improvements might be questioned.

CONCLUSIONS

65. The review process in place at Dounreay, on which the content of this report is based, continues to confirm that safety performance is extremely good.

66. It is also noted that safety issues and the management of safety are regarded at all operational and managerial levels as being of paramount importance.

67. It is noted that ‘industrial safety’ and ‘environmental aspects’ are given equal weight, in the management system and surveillance/response arrangements, to the higher profile ‘radiological safety’.

68. It is stressed that further significant improvements in safety performance can now be achieved by:-

- continuing attention in detail on a very wide front
- attention to the ALARP philosophy
- continued attention to training
- a high level of review associated with information feedback
- and in particular the assimilation and dissemination of lessons learnt from the UNOR system.
ACKNOWLEDGMENTS

69. The input material for this paper arises from sources right across all managerial and support staff within the Fuel Cycle Area. Their indirect contribution to the data and to their responsible operation of the facilities make this report possible.

70. In particular the resources provided by the DTI (and formerly the Department of Energy) by way of funding, and their encouragement is recognised.

71. Particular acknowledgments are given to Mr R H Allardice and Mr O Pugh under whose leadership the PFR reprocessing project was developed and carried through to successful routine operations.
REFERENCES


### Table 1

#### PFR FUEL REPROCESSING

#### ANNUAL THROUGHPUT 1980 - 1994 (to date)

<table>
<thead>
<tr>
<th>Year (nominal)</th>
<th>Reprocessing Plant Operating Days</th>
<th>Throughput Kg HM</th>
<th>Rate Kg/day</th>
</tr>
</thead>
<tbody>
<tr>
<td>1980</td>
<td>92</td>
<td>1173</td>
<td>12.8</td>
</tr>
<tr>
<td>1981</td>
<td>86</td>
<td>956</td>
<td>11.1</td>
</tr>
<tr>
<td>1982</td>
<td>109</td>
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<td>0</td>
</tr>
<tr>
<td>1984</td>
<td>117</td>
<td>1604</td>
<td>13.7</td>
</tr>
<tr>
<td>1985</td>
<td>123</td>
<td>2839</td>
<td>23.1</td>
</tr>
<tr>
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<td>112</td>
<td>2244</td>
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<td>1988</td>
<td>201</td>
<td>2571</td>
<td>12.8</td>
</tr>
<tr>
<td>1989</td>
<td>42</td>
<td>679</td>
<td>16.2</td>
</tr>
<tr>
<td>1990</td>
<td>67</td>
<td>1068</td>
<td>15.9</td>
</tr>
<tr>
<td>1991</td>
<td>115</td>
<td>1804</td>
<td>15.7</td>
</tr>
<tr>
<td>1992</td>
<td>48</td>
<td>363</td>
<td>7.6</td>
</tr>
<tr>
<td>1993</td>
<td>152</td>
<td>1361</td>
<td>8.5</td>
</tr>
<tr>
<td>1994</td>
<td>186</td>
<td>3427</td>
<td>18.4</td>
</tr>
<tr>
<td><strong>Total/Average</strong></td>
<td><strong>1560</strong></td>
<td><strong>23143</strong></td>
<td><strong>14.9</strong></td>
</tr>
</tbody>
</table>

HM ... Heavy Metal (U + Pu)
CLASSIFIED SAFETY RELATED EVENTS IN FRFRP AND ASSOCIATED FACILITIES FROM APRIL 1988 TO MARCH 1994

<table>
<thead>
<tr>
<th>FRFRP</th>
<th>HALS</th>
<th>LABS</th>
<th>SILO</th>
<th>High Radn Store</th>
<th>High alpha Store</th>
<th>LAWP</th>
<th>Total Events</th>
</tr>
</thead>
<tbody>
<tr>
<td>G1</td>
<td>G2+</td>
<td>G1</td>
<td>G2+</td>
<td>G1</td>
<td>G2+</td>
<td>G1</td>
<td>G2+</td>
</tr>
<tr>
<td>Grade Classification</td>
<td>Events</td>
<td>1</td>
<td>2</td>
<td>3</td>
<td>4</td>
<td>5</td>
<td></td>
</tr>
<tr>
<td>----------------------</td>
<td>--------</td>
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<td>---</td>
<td>---</td>
<td>---</td>
<td>---</td>
<td></td>
</tr>
<tr>
<td>GENERIC BASIS OF GRADING</td>
<td>Trivial / low significance</td>
<td>Breach of Good Practice</td>
<td>Regs breach AND/OR reportable injury</td>
<td>Actual or possible serious injury</td>
<td>Actual or possible death or physical disabling</td>
<td></td>
<td></td>
</tr>
<tr>
<td>INES EQUIV.</td>
<td>0-1</td>
<td>0-2</td>
<td>2-3</td>
<td>3-4</td>
<td>5-7</td>
<td></td>
<td></td>
</tr>
<tr>
<td>a - radiation</td>
<td>&lt; 5 mSv</td>
<td>&gt; 5 mSv</td>
<td>&gt; 50 mSv</td>
<td>&gt; 100 mSv</td>
<td>&gt; 2 Gy</td>
<td></td>
<td></td>
</tr>
<tr>
<td>b - Airborne Radioactivity</td>
<td>&lt;200 DACH</td>
<td>&gt;200 DACH</td>
<td>&gt;2000 DACH</td>
<td>&gt;4000 DACH</td>
<td>Major acute effects likely</td>
<td></td>
<td></td>
</tr>
<tr>
<td>c - surface contamination</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>d - personal contamination</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>e - off-site rad’n from Dounreay (CEDE)</td>
<td>&lt;50 uSv</td>
<td>&gt;50 uSv</td>
<td>&gt;500 uSv</td>
<td>&gt;5 mSv</td>
<td>&gt;50 mSv</td>
<td></td>
<td></td>
</tr>
<tr>
<td>f - criticality</td>
<td>Minor error in control terms</td>
<td>Breach of limits</td>
<td>Unintended Criticality</td>
<td></td>
<td>Dose Exposure</td>
<td></td>
<td></td>
</tr>
<tr>
<td>g - Industrial (RIDDOR)</td>
<td>Minor injury</td>
<td>Injury</td>
<td>Stay in Hospital</td>
<td>Injury</td>
<td>Death / loss of limb, etc.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>h - plant damage (involving costs)</td>
<td>Cost &lt; £10K</td>
<td>&gt; £10K</td>
<td>&gt; £100K</td>
<td>&gt; £500K</td>
<td>&gt; £1M</td>
<td></td>
<td></td>
</tr>
<tr>
<td>i - fire damage (involving costs)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>j - explosion (involving costs)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>k - EMITS faults</td>
<td>Minor error</td>
<td>Plant safety not assured</td>
<td>Safety of ops questioned</td>
<td>Unsafe plant conditions</td>
<td>Plant unusable</td>
<td></td>
<td></td>
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<tr>
<td>l - Breach of COSHH</td>
<td>‘Paperwork’ error</td>
<td>Irregularity</td>
<td>Breach of regulations</td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>m - abnormal plant event</td>
<td>Minor anomaly</td>
<td>Shutdown</td>
<td>Oper outside limits</td>
<td>Work outside Rules</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>n - toxic release (on-site)</td>
<td>Mild effects</td>
<td>Exposure &gt; 1 STEL</td>
<td>Exposure &gt; 5 STEL</td>
<td>Exposure &gt; 30 STEL</td>
<td>Exposure &gt; 100 STEL</td>
<td></td>
<td></td>
</tr>
<tr>
<td>o - atmospheric toxic release</td>
<td>Minor release</td>
<td>&gt; 1 x limit</td>
<td>&gt; 10 x limit</td>
<td>Minor effect on local area</td>
<td>Major effect Evacuation</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- **Sv**: Sievert
- **Gy**: Gray
- **DACH**: Derived Air Concentration hours
- **CEDE**: Committed Effective Dose Equivalent
- **s2 of IRRs**: Schedule 2 of the Ionising Radiation Regulations 1985
- **STEL**: Short Term Exposure Limit
- **HRPB**: Highland River Purification Board
- **LT**: Lost Time
- **ALI**: Annual Limit of Intake

39
Fig 1: Sea Discharges from Dounreay

- **Alpha + Beta Activity (Bq)**

- **Year**

- **Activity Discharged to Sea — Statutory Limit**
Fig. 2  Heavy Metal Throughput

<table>
<thead>
<tr>
<th>Year</th>
<th>Total Processed (kg)</th>
<th>Throughput (kg/day)</th>
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</thead>
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<tr>
<td>1980/81</td>
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<tr>
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<td>1984/85</td>
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<td>1986/87</td>
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<tr>
<td>1990/91</td>
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<td></td>
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<tr>
<td>1992/93</td>
<td></td>
<td></td>
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</tbody>
</table>

- Total Heavy Metal Processed
- U+Pu Throughput
Figure 3a and 3b
Figure 4a and 4b
Figure 5a and 5b
Figure 6a and 6b
Serious/disabling injury
Minor injury
Property damage/accident
No injury/no damage/near miss

Bird's distribution of accidents (1969)

FRFR and associated plants events etc.

Figure 7
### Grade 1 UNORs (Events)

<table>
<thead>
<tr>
<th>Year</th>
<th>1: FRFRP</th>
<th>2: HALS</th>
<th>3: LABS</th>
<th>4: SILO</th>
<th>5: High α, β, γ Store</th>
<th>6: High α, low β, γ Store</th>
<th>7: LAWMP</th>
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<tbody>
<tr>
<td>1988</td>
<td>9</td>
<td>1</td>
<td>5</td>
<td>0</td>
<td>0</td>
<td>0</td>
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Facilities:
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2: HALS
3: LABS
4: SILO
5: High α, β, γ Store
6: High α, low β, γ Store
7: LAWMP

Note: Category 1 events only; 1988 and 1994 figures are normalised to 12 months.

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Comparison for all UNORs grades 1-5

Facilities
1: FRFRP
2: HALS
3: LABS
4: SILO
5: High α-β+γ Store
6: High α- low β-γ Store
7: LAW

NOTE: Data for 1988 and 1994 are normalised for 12 months

Figure 9
OPERATIONAL EXPERIENCE AT THE TOKAI REPROCESSING PLANT

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Abstract

The Tokai Reprocessing Plant (TRP) is the first reprocessing plant with a capacity of 0.7 t per day as an industrial scale in Japan. Power Reactor and Nuclear Fuel Development Corporation (PNC) has been operating the plant since 1977. The total amount of reprocessed fuel is about 756 tonnes including 5 tonnes of MOX fuel from Advanced Thermal Reactor (ATR) "FUGEN" by the end of June in 1994.

This paper summarizes mainly an operational experience gained at TRP after the latest meeting.

1. Present Status of TRP

1.1 History of TRP

The reprocessing project of PNC was started when the Atomic Energy Commission (AEC) of Japan decided that reprocessing of spent fuel and treatment of radioactive waste should mainly be done by the Atomic Fuel Corporation (AFC) in September 1956. AFC is the predecessor of PNC.

AFC entered into a contract for a preliminary design of the plant with the Nuclear Chemical Plant (NCP) of UK in 1963. A detailed design was started by the Societe Generale pour les Techniques Nouvelles (SGN) of France in 1966.

The permission for plant construction was granted by the Japanese Government in 1970 and the construction was started as a joint venture of SGN-JGC of Japan in 1971. The Plant was completed in 1974 and hot test started in September 22, 1977 after completing the U test using unirradiated uranium. The operational license was granted after passing the governmental inspections by the end of 1980. Regular operation has been started since January 1981. (See Figure 1)

PNC has obtained valuable experiences for LWRs and ATR MOX fuels reprocessing through the operations.
1.2 Amount of Reprocessed Fuel

The total reprocessed fuels between the start of hot operation in September 1977 and the end of June in 1994 is about 755.5 tonnes of oxide spent fuel. (See figure 2) The kinds of spent fuel assembly are BWR, PWR and ATR "Fugen" Mixed Oxide (MOX) fuel. The numbers of the assembly and the amount of spent fuel are as follows, BWR: 2,440 assemblies (454.0 tonnes), PWR: 727 assemblies (266.5 tonnes), ATR: 170 assemblies (26.1 tonnes) including 34 assemblies (5.2 tonnes) of MOX. (See Table 1)

The amount of plutonium nitrate recovered as a final product was about 5.1 tonnes. Most of the recovered plutonium has already been sent to Pu conversion plant for fabricating the MOX fuels to the ATR "Fugen", the experimental FBR "Joyo", and proto-type FBR "Monju".

1.3 Major Maintenance Activity and Scheduled Shutdown of Plant Operation

TRP experienced major maintenance activities such as replacements and modifications of main equipments. They interrupted the continuous operation of whole process. The operation became steady and stable since 1985 after these activities performed.

However, it is required to increase the reprocessing amount more than before, because of the demand for more plutonium for ATR and FBR fuel cycle development. Therefore, scheduled shutdown of plant operation was set up and improvements and modifications of main equipments and major process has been carried out to prevent the failure of major equipments due to corrosion and etc. As a result, the annual operation days were increased and the performance factor, operation rate was improved.

Improvements and modifications of main equipments and major process are as follows.

(1) Remote Repair and Installation of Dissolvers

There found pin holes in the welded part on the barrel of dissolver. The remote repair technology had been developed, and remote in-situ repair of dissolver R10 and R11 was carried out from September to November 1983. A new dissolver R12 was fabricated with improved material and no welded lines is in the inside steam jacket. A fabrication of dissolver was finished in April 1984, and was installed by the end of November 1984.

(2) Repair and replacement of Acid Recovery Evaporator

There found pin holes in welded part of heating tube in the acid recovery evaporator after 6000 hours operation, so exchange of whole part of evaporator was done by end of December 1979. However, the second one leaked again after 13,000 hours operation in February 1983 and only boiler part of evaporator was replaced with domestic produced materials, which was 25%–chromium and 20%-nickel alloy of stainless steel. The decision was taken to replace the third evaporator as a preventive maintenance with the new one made of Ti-5%Ta alloy in June 1988.
(3) Replacement of Plutonium Solution Evaporator

The design of original plutonium evaporator was to connect the washing column to boiler part with the flange. After 10,000 hours of operation in year 1982 in-situ repair was done and in year 1984 the replacement of whole evaporator was done after 12,000 hours of operation. Again the replacement of the evaporator was done after 9,000 hours of operation.

(4) Modification of Boiler part of Acid Recovery Distillator

The part of heating coil of the acid recovery distillator was repaired in February 1981 after 13,000 hours of operation. In 1984 the boiler part of distillator was replaced. The new distillator was installed to replace old one after 13,000 hours of operation.

(5) Modification of Fuel Assembly Shearing Machine

Many modification works for internal parts of shearing machine were done to improve the operability and maintenance ability.

(6) Installation of Second Pulsed Filter

To improve the plant efficiency factor, second pulsed filter was installed in the clarification process. The new type of valve for changing connection of both filters was developed to install inside cell for easy maintenance and high fidelity.

1.4 Radioactive Waste Discharge

In the normal operation of TRP, low level radioactive effluent is discharged to the atmosphere and the sea under rigid control. Radiation exposure of the public around the plant has been estimated for the potential pathways with the site specific parameters such as food consumption, concentration factors of marine organisms and meteorological condition.

External exposure due to gamma ray from 85Kr and internal exposure via inhalation and oral intake of radio-nuclides are evaluated for the airborne effluent. External exposures from contaminated fishing net and fishing boat are considered as pathways for fishermen. External exposure to contaminated beach and internal exposure via oral intake of marine products are evaluated in connection with the liquid effluent.

Estimated annual effective dose equivalents are only less than 0.1 percent of the annual effective dose equivalent limit for the public recommended by the International Commission on Radiological Protection (ICRP) since hot operation in 1977.

Approximately 542 m3 of Highly Active Liquid Waste (HALW) had been generated by the end of March 1993. It has been stored in stainless steel tanks. The annual total $\beta$ activity discharged into the sea, was now only 10-2 GBq order and the annual I-129 discharged to the atmosphere was 10-1 GBq order. (See Figure 3)
1.5 Radiation Exposure Control of Plant Personnel

Radiation control is based on the authorized regulation in Japan and the ALARA principle at TRP. Occupational exposure is limited in the regulations, i.e. effective dose equivalent limit of 50 mSv in a year. It is necessary to prepare a special radiation work plan when the exposure rate of an operator is expected to exceed 1 mSv per week. To minimize exposure and avoid excessive exposure of an individual in the plant, investigation levels for exposure are set over three months, for instance, 3.7 mSv for effective dose equivalent. If a person receives 3.7 mSv per 3 months, an investigation must be carried out to determine the cause of the exposure. Exposure rates and concentrations of airborne radioactive materials are measured continuously by the automated monitoring system. Through this strict radiation control approach, the exposure rate of personnel is kept at a low level. (See Figure 4)

The annual exposure of personnel was about 1.3 man Sv from 1992 to 1993 (Fiscal year).

2. Outline of Recent Event

2.1 Radiation Exposure of Workers (see Annex)

Four workers were exposed internally while they were replacing a filter element of the vacuum filter on December 27, 1993. It is installed in the sampling system connected to the plutonium receiving vessel of the rework process. At the time of incident, the operation of the Plant was stopped for routine maintenance from December 1993.

The cause of the internal exposure was that the four workers inhaled plutonium particle dispersed from the filter element in the vacuum distribution room. The floor of the room was contaminated by this radioactive release.

The bioassay analysis results showed that the maximum estimated internal exposure over in 50 years was 90 mSv of effective dose equivalent and 1700 mSv of tissue dose equivalent for one of 4 workers. The both of dose equivalents exceeded the legal dose limit (50 mSv per year and 500 mSv per year), respectively.

2.2 Failure of Spent Fuel Supply Conveyor

A shearing operation was interrupted by the failure of spent fuel supply conveyor occurred on April 12, 1994. This event was the minor event that is not necessarily report to the STA legally.

As a result of the observation in the cell, it was found out that the driving shaft of the supply conveyor was out of joint on the side of driving motor unit. The shaft is connected with insert shaft of driving motor unit and gear box. The cause of the incident was that a setscrew was loosened and had not act. The setscrew is to prevent coming the driving shaft off from the insert shaft.
Driving shaft was replaced by new one with insert shaft. The setscrew was fixed by spot welding. Repair works were finished on April 15, 1994 and the plant operation was started.

2.3 Suspension of Compressor by Electric Power Loss

There was a main electric power loss due to lightning struck in the vicinity of the Tokai Works on May 8, 1994. This event was the minor event that is not necessarily report to the STA legally.

Most of the main process equipment was stopped for a moment and the emergency power supply unit went into operation automatically at TRP. After 7 seconds, main power supply made a quick recovery from power loss, but the compressors for measuring instruments were suspended owing to power supply cutting off by the circuit breaker. After 1 hour, the compressors were recovered. Some measuring instruments were held up for 1 - 3 hours. However, TRP maintained in safety condition.

Detailed investigation after the suspension of compressor revealed the findings as follows. TRP has two power distribution bus bars which are provided with emergency power supply units respectively against the main electric power loss. It happened that one bus bar was supplied with the main power and the another with the emergency power at the main power recovery. The cause of power supply loss by the circuit breaker is that the two different electric currents were connected unexpectedly to provoke the over current at the power supply circuit of the compressors.

That is attributed to two inadequacies. The one is that the check list of automatic cross bar switch that connects two bus bars against the bus bar fault was inadequate. Another one is that the cross bar switch was not provided with interlock system that prevents the different electric currents from connecting.

As a countermeasure, the improvements on check lists of the switching boards were taken and the interlock system was installed on the cross bar switch.
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**Figure 1: History of TRP**
### Table 1 Total Reprocessed Fuel at TRP

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Figure 3 Activity Discharge
Figure 4  Radiation Exposure at TRP
ANNEX

1. Event title

Internal exposure of workers by plutonium at the Tokai Reprocessing Plant.

2. Outline of the event

On Monday, 27 December 1993 workers who replaced a vacuum filter (see Figure 4) which is in the vacuum sampling line (see Figure 3) from the plutonium solution receiving tank [276V20], a part of the reworking process inhaled plutonium. The replacement of the filter was done in the vacuum distribution room [A684] and airlock [A685] (see Figures 1 and 2) at the main building of Tokai Reprocessing Plant (TRP). When the workers survey each other after they replaced the vacuum filter, contamination was found on the clothes of the workers and in the work area. This paper describes the event itself, evaluations of the internal dose of workers, causes of the contamination, and countermeasures to be taken to prevent similar accidents in the future. This event was the minor event that is not necessarily report to the STA legally.

3. Evaluations of internal dose

From the results of bio-assays on the four workers, the effective dose of workers A, B, C and D were exposed 16 mSv, 90 mSv, 2.6 mSv, and 5.1 mSv respectively (dose limits of 50 mSv/year); dose of the bone surface 290 mSv, 1700 mSv, 48 mSv, and 95 mSv respectively (against allowed limits of 500 mSv/year). The effective dose of the four workers of the past one year are 0.3 mSv, 1.2 mSv, 0.2 mSv, and 0.7 mSv respectively.

In evaluating the dose of the workers, we assumed that the plutonium was in the form of nitric oxide and we followed the procedure of the Science and Technology Agency.

4. Causes (see Table 1 for details)

(1) Events leading to the release of radioactive material

The replacement of the vacuum-system filter was done in the following procedure.

1) On the afternoon of December 24, worker A and worker B placed a plastic sheet over the area surrounding the filter in the vacuum distribution room [A684].

2) Workers A and B put on half face masks covering the lower half of their faces.

3) Workers B and D together lifted up the filter from its casing with the cover attached.

4) While worker D held the filter with the cover attached, worker B wrapped the filter by a wet towel and then removed the cover from the filter.

5) Worker B then used a single plastic sheet to wrap the filter but not sealed completely.

6) Worker B then handed the filter to worker C. He measures the radiation dose of the filter but did not measure the surface concentration.
7) Worker A then wrapped the filter by a second plastic sheet in the airlock [A685], after that he moved the filter to another vacuum distribution room [A682].

8) Worker D removed the O-ring remaining in the casing and installed a new filter.

9) After the completion of the work, the surface concentration was found on the surface of the surrounding area.

10) Internal contamination was found in nasal smear of workers B and D, A C as well.

11) Since contamination of a level of 33 Bq/cm² had been found in the work area (i.e., A685), and designated as a restricted-entry area

2) Causes of radioactive material release into the vacuum distribution room [A684].

   From the following reasons, the cause of the release of radioactive material that causes the exposure of the four workers is estimated as in the process of replacing the vacuum filter in the vacuum distribution room [A684].

1) The vacuum distribution room [A684] is a controlled area and entry of workers into the room is normally restricted; there is thus no way of contamination brought outside, and there was no contamination observed before the occurrence of the current accident.

2) The results of measurements taken using the air sniffer installed in the vacuum distribution room [A684] show that high level during the period of replacement.

3) No sampling operation related to the vacuum filter in the vacuum distribution room [A684] was performed.

4) The characteristics of the nuclides attached on the inside of the vacuum filter almost the same as the nuclides found in the bio-assays.measurement.

5) Contamination (of a level of 11 Bq/cm²) was found on the outer surface of the first plastic sheet used to wrap the filter.

(3). Causes of the internal exposure:

1) The cause of the internal exposure of workers B and D lies in the fact that they were exposed to the air that had been contaminated as a result of the work done in wrapped the filter that causes the spread of radioactive contamination in the vacuum distribution room [A684].

5. Measures taken to prevent accident

1) Measures to be taken to prevent a reoccurrence of the release of radioactive material during the replacement of vacuum filters

   1) In order to preventing the spread of radioactive material, the open-area replacement procedures that have been used partly until this time changed to the sealed replacement procedures that will eliminate direct external contact.
2) Manual is revised in accordance with the change to the use of sealed replacement procedures. And while checkpoints for radiation survey are not explicitly noted in the manuals currently in use, the revised manuals will make explicit note the checkpoints to check radiation survey.

Training of workers using these manuals will be conducted under the supervision of experienced personnel in the work place and under conditions matching those of actual working conditions.

(2) Preventing similar accidents
In order to prevent the occurrence of similar accidents and in consideration of the need to prevent the spread of radioactive material, manuals on working with non-sealed plutonium and plutonium are reviewed in accordance with the views noted above in (1), workers will be trained on the importance of strict adherence to manual procedures.

(3) Caution in planning
For the higher levels of safety when performing tasks where workers may be exposed to radioactivity, a new radioactivity safety checklist are installed in addition to the general safety checklist now in use so that workers will be able to perform sufficient preliminary checking to make them even more aware of the hidden dangers to which they may be exposed in their work and to thereby prevent them for overlooking anything of significance. The role of the site foreman with respect to ensuring safety and monitoring the progress of work will also be explicitly described within the work procedures.

(4) Strict adherence to safety regulations
The procedures followed at the time of the current accident were in some respects inappropriate in regard to the actions taken with respect to the requirements outlined in the safety requirements that require the notification before moving nuclear materials and other radioactive materials. So the measures that are to be taken to prevent the spread of contamination during the work, and checking of contamination and other steps that are to be taken to protect workers from exposure are clearly written in the safety regulations. These safety regulations are strictly observed at all times.

6. Decontamination of the room

(See Figures 6 and 7)
After the contamination had occurred, preventive measures are taken to control the spread of contamination, afterward the room was decontaminated on December 28. Contamination of the room fell below the value of 0.04 Bq/cm².

7. Conditions of the exposed workers

Immediately after the accident occurred, all four workers were examined by a physician who determined that there was nothing wrong with their health.

At the present time he has no noticeable problems with his health, and he continues to work in non-controlled areas.

Attached materials

Figure 1: Layout of the 6th floor of the separation and refinement plant
Figure 2: Diagram of work area
Figure 3: Sampling system
Figure 4: Vacuum filter
Figure 5: Spread of dust within the vacuum distribution room
Figure 6: Distribution of radiation
Figure 7: Distribution of radiation after decontamination
Table 1: Comparison of work procedures against actual work performed
Figure 1. Layout of the 6th floor of the main plant

[Diagram showing the layout of the 6th floor of the main plant with labeled areas such as Work area, A683 airlock, and Amber area.]

[Legend: Green area]
Figure 3. Sampling system
Figure 4. Vacuum filter
OECD/NEA TOPICAL MEETING
SAFETY OF THE NUCLEAR FUEL CYCLE
Cadarache, September 20-21, 1994

Tomsk-7 Nuclear Event
Causes, Consequences And Lessons Learned

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Summary: The paper provides general information on main technological causes of the accident which had happened at Tomsk-7 radiochemical plant on April 6, 1993. The technological operations during the event, human faults, control system inefficiency and consequences of the accident are reported. Information is given on lessons learnt for safety improvement of Minatom's radiochemical facilities.

1. Overview of some radiological accidents

The radiochemical plants design must include multibarrier protection and safety systems in order to prevent environmental release of radioactive materials above the permitted limits.

The majority of known and/or reported nuclear accidents of chemical nature was caused by:
- Process criticality (e.g. solution overconcentration);
- Ignition and/or explosion of different inorganic materials;
- Ignition and/or explosion of organic materials (ion-exchange resins, extractants, etc), including "red-oil" type reactions.

The last two causes could be considered the result of an uncontrolled chemical reaction. One of the most serious nuclear event of a chemical nature has happened in 1957 at Production Association "Naryak" (Chelyabinsk-55) radiochemical plant. The accident resulted from the dry nitrates & acetate salts explosion because of the cooling and control systems failure in a high level liquid waste storage tank. As the result of the event about 20MCi of radioactive wastes had been ejected from the vessel. 2MCi of these radioactive materials have spread for a 100 km beyond the site border [1].

A number of different accidents occurred in US radiochemical processing facilities [2]. A large fire happened at the Rocky Flats plant in 1969 as the result of metallic plutonium ignition. The fire was aggravated by neutron shields combustion. In 1949 an explosion of a nitric acid-hexone system occurred in Hanford. This resulted in heavy damage of the pilot facility. Several "red oil type" reactions have occurred at the US nuclear sites: Savannah-River, Hanford, Oak Ridge. The last incident of chemical nature in the USA has happened in 1975 [2,3].

Some uncontrolled chemical reactions involving ion exchange columns should be mentioned: the event of plutonium-238 exchange column rupture in 1993 at PA "Naryak" radioisotope production plant is similar to some extent to the americium ion exchange column explosion in 1976 at Hanford [2].

One can consider the Tomsk-7 case as a very typical nuclear event at radiochemical processing plant.

Nevertheless the incident attracted attention to the ge-
neral problems of reprocessing safety because of the plant damage and off-site environmental risk.

2. Tomsk-7 nuclear site

Minatom's Siberian Chemical Combined Works (SChCW) is well known as Tomsk-7 nuclear site and is located few tens of kilometers to the northwest of the city of Tomsk in Western Siberia. The site combines several different nuclear and civil facilities, including:

- two dual-purpose uranium-graphite reactors (other three old production reactors were shut-down in 1990-92);
- uranium enrichment plant;
- radiochemical plant for metal uranium spent fuel reprocessing;
- civil pure materials & chemical production facilities, as well as various auxiliary technological and waste management facilities.

First nuclear reactors were built at the site in the 1950s. Reprocessing plant is a double canyon buildings. Original design of the plant was based on uranyl acetate precipitation technology. In 1983 the plant was reconstructed for PUREX process with nitric acid - 30% TEP in paraffin hydrocarbon diluent extraction system. For the purpose one of the plant's canyon was made deeper and 5 pulsed extraction columns were installed. Some of the apparatuses and vessels were not changed. These pieces of equipment were adapted for the PUREX technology. Liquid waste solutions treatment includes additional U and Pu recovery with precipitation, sludge dissolution and periodical solvent extraction. Deep well injection is used up to now for the final disposal of liquid radioactive wastes.

3. Description of Tomsk-7 event

On April 6, 1993 at 12.58 local time at Tomsk-7 radiochemical plant during the acidity adjustment of the uranium solution for further extraction purification a rupture of a technological vessel in which adjustment operations were carried out has happened, as a result of a rapid pressure increase.

Ejected gas-aerosol mixture blew away concrete cover of the cell and exploded in the above-the-canyon maintenance corridor. The mixture ejection and explosion damaged section of the roof above the cell, blew out a part of masonry wall and glazing. As a result of the event some radioactivity released into the environment through stack and openings in the building.

An exothermic chemical reaction occurred in the stainless steel cylindrical Process tank AD-6102/2. Total volume of the tank was 54.15 cbm. The tank was installed in a reinforced concrete canyon 10.4 m deep with stainless steel lining. The canyon volume was about 100 cbm. Dimensions of the tank and the canyon's sell are given in Fig.1. This tank was used for solution's technological adjustment for temperature, composition and acidity before the next processing stage of solvent extraction purification.

AD-6202/2 apparatus was equipped with necessary remote control and operating systems. Control systems included temperature gauge, pressure transducers, overflow indicator, upper and lower level alarms and some others. Solutions were mixed by air spargers. The tank was equipped with heating and/or cooling water jacket, covering about 40% of the bottom part of the tank.

2.5 hours before the accident 1.5 cbm of concentrated
nitric, (14.2 M) acid has been added to the 23.5 cbm of uranyl nitrate solution transferred to the tank previously. The uranium solution was composed of three batches:

4.0 cbm of bottom uranium solution, remained in the tank after finishing of the previous operations.

19.5 cbm (in two portions) of uranium solution concentrated by evaporation. This solution contained some of Pu after the first purification cycle, 440 grams U/L and 0.5 M nitric acid.

The resulting uranium solution in the tank contained 320 grams of Pu, 6773 Kq of uranium, 0.5 M nitric acid, 550 Ci of total radioactivity.

At 12.45 the emission of gaseous nitrogen oxides as a stack fume was noticed, soon after that pressure increase in the tank AD-6102/2 was detected. At 12.55 pressure in the tank has increased to 5 atm and the pressure increase continued in spite of the measures taken. At 12.58 the tank exploded.

Estimated release of radioactivity was 10% of the tank content.

According to the International Nuclear Event Scale developed by IAEA this event was qualified as a 3-d level event - "serious incident".

Beyond the enterprise boarder the contaminated area was 26 km long, area with a gamma-radiation level of 15 mRem/hour and higher was 123 sq. km.

Original isotope composition of the contamination was:

- Zr-95 + Nb-95 - 62% (T 1/2 = 64d)
- Ru-106 - 35% (T 1/2 = 1.02a)
- Ru-103 - 2% (T 1/2 = 39.3d)
- Cs-137/Sr-90 - <1%

Thanks to the snow-fall during the event major part of the radioactivity fell out close to the plant buildings. The contaminated snow was timely collected at a special place and later on the resulted wastes were buried.

Decontamination and repair works at the plant took 4 months.

As a result of the event investigation by the Minatom’s scientific team it was ascertained that an exothermic chemical reaction has happened between an organic compounds and concentrated nitric acid. The organic phase quantity in the tank was estimated as 0.2 cbm. Some organics were found and analysed after the accident, the TRP content in the organics was 26%.

Several causes for the starting of the oxidation process and fast overpressure were pointed out:

- evident lack of sparging in the tank after nitric acid was transferred;
- presence of degraded organic matter in unreasonable amounts in the tank due to deficiency of organic phase control system;
- deficiency of temperature control in the upper part of the vessel; that is, the thermocouple was installed at the bottom part of the tank at the mark of 1.4 cbm only;
- the control vent valve was open only to 70% of its area;
- 14.2 M nitric acid was used instead of 12 M acid limited by process documentation.
The consequences of the rapid chemical reaction were aggravated by the fact that the tank could withstand very high internal pressure up to 20 atm and was not equipped with large relief valves.

4. Lessons learned after the Tomsk-7 event

To avoid similar incidents at Tomsk-7 and other Minatom's radiochemical facilities a number of recommendations were developed and mostly put into practice.

- Limiting the concentration of nitric acid that is used for solutions adjustment to not more than 6M.

- Implementation of reliable interlock system to prevent addition of nitric acid without proper mixing/sparging of the solution.

- Improvement of the temperature control system in the large/high vessels.
  - Limiting the tank solution temperature of less than 70°C

- Providing visible alarm signal to the personnel when technological limits are exceeded. Additional personnel training was conducted for the case of emergency.

- Improvement for liquid waste solutions treatment to prevent mixing of head-end and back-end solutions.

- Improvement of organic compounds control in the tanks, development of a reliable direct remote control of organics in the highly radioactive solutions.

- Maintaining constant ventilating of the tanks to prevent hydrogen accumulation.

After the Tomsk-7 event became evident inefficiency of a present environment off-site monitoring system. Necessary improvements were made for a rapid communication in case of emergency.

Special 1994-95 R&D program was developed for Nuclear and Radiochemical Safety improvements of Minatom's enterprises.
References


GESTION DES DOSES EXTERNES ET DU RISQUE DE CONTAMINATION DANS L'USINE BELGONUCLEAIRE DE FABRICATION DE COMBUSTIBLE MOX A DESSEL

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RÉSUMÉ :

Dans ses installations situées à Dessel, BELGONUCLEAIRE s'emploie à la fabrication de combustible MOX pour réacteurs à eau légère. Depuis 1989 la production y a atteint son niveau nominal à 35 tonnes de combustible par an, pour une occupation d'environ 250 personnes.

La chaîne de production se compose d'opérations de mélange des poudres d'oxyde d'uranium et de plutonium, de pastillage, de rectification et d'engainage, l'ensemble de ces opérations étant effectuées en boîte à gants. Les crayons soudés scellés sont ensuite soumis à un programme de contrôle non destructif, avant d'être expédiés à l'usine voisine de FBRF pour mise en assemblage.

Les problèmes spécifiques de sécurité et de sûreté que pose la mise en œuvre du plutonium nécessitent une attention particulière en ce qui concerne la gestion des doses externes et du risque d'incident de contamination dans les installations.

Les doses externes sont essentiellement engendrées par le travail continu dans des champs de débits de dose relativement faibles. Elles se discrènent en doses photons et doses neutrons.

Dans l’optique ALARA, des efforts substantiels de réduction des doses ont été, et sont entrepris, avec un accent particulier sur les mesures de protection collectives : conduite à distance de la chaîne de production, blindage systématique des postes de travail, sensibilisation du personnel.

Ces efforts ont d'ores et déjà porté leurs fruits en ce qui concerne les doses photons, la dose collective photons ayant pu être réduite progressivement de 1050 mSv en 1989 à 650 mSv en 1993.

Les doses neutrons sont aujourd'hui pratiquement égales aux doses photons, soit approximativement 600 mSv. Cette situation est attributable d'une part à la difficulté de mise en œuvre de blindages neutrons, d'autre part aux problèmes de précision de la dosimétrie neutrons et à l'augmentation sensible du taux de plutonium 238, principal générateur de neutrons.

Les efforts futurs de réduction de doses seront dès lors axés sur les postes de travail à débit de dose neutron élevé.
La maîtrise du risque de contamination nécessite une surveillance rigoureuse et permanente des installations. La norme très restrictive de contamination surfacique maximale admissible, à 0.5 Bq/dm², permet de garantir un état de propreté essentielle de la zone contrôlée. De plus, dès qu’une contamination est décelée, des actions palliatives immédiates sont prises.

L'expérience montre que les contaminations, causées par perte d'étanchéité du confinement, sont presque exclusivement attribuables au percement des gants utilisés en boîtes à gants. Un programme de contrôle a été mis sur pied, prévoyant le contrôle systématique des gants à la réception, et un contrôle périodique de l'ensemble des gants installés sur boîtes à gants. Ces actions ont permis de réduire sensiblement le nombre d'incidents de perte d'étanchéité.

Les efforts actuels portent quant à eux sur la recherche de moyens améliorant la protection des gants, et sur une plus grande sensibilisation du personnel travaillant en boîtes à gants.
A Programme for the Assessment of Radiological Risk Associated with Active Handling Operations

by

David Iain Hambley

Summary

This paper describes a risk assessment programme which has been developed for the evaluation of radiological risk posed by non-reactor nuclear plants within the United Kingdom. The risk assessment parameters and thresholds of acceptability, relevant to plants located on nuclear licensed sites within the UK, are identified.

The structure of Probabilistic Safety Assessments (PSAs) for nuclear reactors is examined and compared with manual methods used for non-reactor plant. The chosen methodology and scope of the programme are derived from this comparison. The source of the main dispersion equations and the methods used to derive the main risk assessment parameters are described. A rudimentary sensitivity analysis, implemented in the programme, is described and compared to sensitivity analyses developed for nuclear reactor PSAs.

Recommendations have been made for further work, which would lead to a living PSA tool for use on plants handling radioactive materials.

To be presented at a topical meeting on the 'Safety of the Nuclear Fuel Cycle', sponsored by the OECD Committee on the Safety of Nuclear Installations, at Cadarache, France 20-21 September 1994.

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August 1994
Introduction

This paper describes the development of a computer programme for the assessment of risks, to the public and to workers, posed by plants which handle radioactive materials. The assessment methodology used within the programme is shown to comply with the requirements of the United Kingdom regulatory authorities and AEA Technology.

1.1 REQUIREMENTS FOR RISK ASSESSMENT IN THE NUCLEAR INDUSTRY

The operators of nuclear plant in the United Kingdom are subject to the requirements of the Health and Safety at Work Act \(^{20}\), in common with all places of work. Thus they are required, inter alia,

"to ensure, so far as is reasonably practicable, the health and safety of all their employees"

and

"to conduct their undertakings in a way that ensures, so far as is reasonably practicable, that persons not in their employment who may be affected are not thereby exposed to risks to their health or safety."

This means that measures necessary to avert risk must be taken unless the costs of those measures become disproportionate to the risk that would be averted by their introduction. The requirement that risks should be reduced to a level which is as low as reasonably practicable is usually called the ALARP principle.

Operators of nuclear plant must also comply with the Nuclear Installations Act \(^{20}\), which is subsidiary to the Health and Safety at Work Act. Under this act, apart from certain exceptions, no site may be used for the purposes of installing or operating any nuclear installation, unless a licence has been granted by the Health and Safety Executive (HSE). The Nuclear Installations Inspectorate (NII), which is part of the Nuclear Safety Division of HSE, are responsible for the granting of licenses and for attaching appropriate conditions, where necessary. The NII are also responsible for making judgements on the acceptability of the responses made by the licensee to the requirements of the licence and any conditions.

A key part of the process by which the NII decide whether to grant a particular Site Licence is the assessment of safety cases provided by the licensee. The aim of the safety cases is to justify continued operation of the plants on the site. Safety cases are also required to demonstrate that a proposed modification does not produce unacceptable risks. Each safety case must demonstrate that, inter alia,

- the hazards posed by the normal operation of the plant are acceptable,
- the risks posed by conceivable fault sequences are acceptable,
- the management systems within each plant are adequate to control normal and accident conditions,
- the plant is being adequately maintained in the light of the potential hazards,
- the plant conforms to modern standards of engineering or, for existing plant, that the risks posed by plant designs which do not conform to modern standards of engineering are as low as reasonably practicable.

It is generally accepted that such a Safety Case requires a probabilistic assessment of the risk to the public and the workforce, since for most plant deterministic safety cannot be proven.

1.2 STANDARDS OF RISK ASSESSMENT

HSE published the NII's safety assessment principles (SAPs) for nuclear power plants in 1979\(^{48}\) and for nuclear chemical plant in 1983\(^{20}\). In 1988, HSE published an updated SAP document for nuclear reactors\(^{48}\) and a discussion document on the tolerability of risks from nuclear power stations\(^{49}\).

In 1993 the HSE published revised Safety Assessment Principles\(^{48}\), covering both nuclear reactors and nuclear chemical plants, and a revised version of the report on tolerability of risk from nuclear power stations\(^{49}\). These reflected developments in risk assessment methods which had occurred since publication of the previous versions of the SAPs, comments made at two public inquiries into proposals to build nuclear power stations at Sizewell\(^{48}\) and Hinkley Point\(^{48}\), and lessons learned by the NII in the intervening period. The publication of these documents was concurrent with the Royal Society's revision of its own report on

1 AEA Technology

77
risk assessment\(^{17}\).

The tolerability paper\(^{16}\) provided a justification for the levels of risk from nuclear plants which could be considered as 'tolerable'. This justification took account of perceived and actual measures of tolerable risk and of the 'dread factor' associated with nuclear plants. The revised SAPs\(^{20}\) specify detailed risk thresholds based on these 'tolerable' levels of risk. In addition, they require the assessment of uncertainties in the risk analysis and the quantification of the effects of a modification on the risks posed by the plant. The SAPs also list design principles which, effectively, define what the NII consider to be modern standards of engineering design.

1.3 EXISTING COMPUTATIONAL METHODS

The regulatory requirements and the complexity of nuclear plant have led to the development of computational methods for risk assessment within the nuclear industry. For nuclear power plants these methods have been developed over many years and have now reached a high level of sophistication\(^{6,10}\). Large suites of inter-connected computer programmes can carry out risk assessments and sensitivity analyses for all fault conditions for nuclear power reactors. Programmes have also been developed to provide real time risk values based on current plant conditions\(^{27,18}\).

By comparison, risk assessment methods for non-reactor nuclear plant rely heavily on manual methods\(^{9}\), which cannot readily assess the sensitivity of the resulting risks. Computer programmes used in risk assessments for non-reactor plant apply in general to single stages, e.g. to evaluate fault trees\(^{20}\). Therefore a need was identified to provide a risk assessment programme, similar to those used for nuclear power reactors, applicable specifically to non-reactor nuclear plants.

In this paper active handling plants are considered to cover all non-reactor nuclear plants. Plants designed for long term storage of radioactive wastes and treatment of radioactive liquids are not covered by this definition of active handling plants.

2.0 Risk Assessment Parameters

The broad requirements for radiological risk assessment have been laid out in the introduction. These requirements are now described in more detail.

2.1 REQUIREMENTS OF THE ASSESSMENT METHODOLOGY

The NII require\(^{16}\) that the PSA for a plant should consider all initiating events identified by whatever means are considered appropriate by the operator. Thereafter account should be taken of component failures, component unavailabilities due to maintenance or testing, common cause event and personnel errors [Principles 16 & 33].

The PSA should also provide information on the reliability, maintenance and testing of safety systems [Principle 41]. The sensitivity of PSA results to input data, assumptions and calculation methods must also be assessed [Principle 52].

The PSA results must be subjected to an independent check [Principle 53], therefore the results of the PSA must be sufficiently clear to allow such a verification.

The fault analysis must be reviewed and if necessary updated during the life of the plant. Updates may be required because of changes to the plant or operations, or as the result of information gathered during plant operation [Principle 54].

2.2 NUMERICAL RISK CRITERIA

Two sets of numerical criteria are given in the NII SAPs\(^{40}\): a Basic Safety Level (BSL) and a Basic Safety Objective (BSO). The BSL is the maximum tolerable dose/frequency/risk. The BSO is the dose/frequency/
risk threshold below which detailed ALARP justification need not be made. The BSL and BSO are typically two orders of magnitude apart.

The primary risk criteria for doses to the public are defined in terms of a dose frequency staircase [Principle 42] as shown in Figure 1. There is also a requirement to ensure that no single class of accident exceeds more than one tenth of the total frequency in any dose band.

The primary risk criterion for dose uptake by the workforce is that the individual risk of death from all accidents should be less than $10^{4} \text{ y}^{-1}$ to meet the BSL and less than $10^{4} \text{ y}^{-1}$ to meet the BSO.

There is an additional criterion related to a 'large release' [Principle 44]. A 'large release' is defined as greater than 10,000 TBq of Iodine 131 or 200 TBq of Caesium 137 or a mixture of other isotopes which would give rise to similar consequences. The BSL frequency for a large release is $10^{-3} \text{ y}^{-1}$ and that for the BSO is $10^{-7} \text{ y}^{-1}$.

The programme does not seek to address the assessment principle relating to plant damage [Principle 45] because of the degree of interpretation required to define what constitutes 'plant damage' in the context of any given active handling plant.

### 2.3 ADDITIONAL REQUIREMENTS

For the purposes of categorising the potential hazards associated with a modification to the plant, UKAEA Corporate Safety Instruction No 80/46 requires that the maximum possible dose uptake to the public is calculated. This is in contrast to the best estimate doses which are used for comparison with the NII SAP criteria.

Another parameter which is often of interest on nuclear sites is the frequency with which the lower Emergency Reference Level (ERL) is breached. The Emergency Reference Level is the short term dose which someone situated at the site boundary could notionally receive before action would be taken to protect people outside the site.

### 3.0 A Generic Model of the Accident Scenario

This section describes the general philosophy behind the model of the accident scenario which is used in the assessment programme.

### 3.1 A GENERIC INCIDENT SCENARIO MODEL

A generic model for the way in which an incident develops into an accident is presented in Reference 23. In this model an incident is defined as "all undesired circumstances and near misses which have the potential to cause accidents." An accident is defined as "any undesired circumstances which give rise to ill health or injury; damage to property, plant products or the environment; production loss or increased liabilities."

The referenced model postulates that an incident begins with an initiating event (or immediate cause), although other root causes within the sociotechnical system of the company or plant will also be present. The effect of the initiating event is to place the plant in a disturbed condition. If the situation is not controlled then the plant will pass into a hazardous condition. The incident is then postulated to progress through time via a sequence of steadily worsening plant states. Between each plant state opportunities exist to eliminate or mitigate the progression of the incident. An accident condition occurs because of failure to control or mitigate the effects of the incident. Thereafter further escalation of the accident can be modelled similarly.

### 3.2 MITIGATION

A key concept in the model described above is that of mitigation. Mitigation or control can be exercised in
a number of ways during the progression of an incident. The most obvious methods of mitigation are installed physical controls which act to negate or reduce the effects of a release. Examples of this form of passive physical mitigation include filters and containment barriers. The progression of an incident can also be controlled by the action of operators and control systems. This form of mitigation is reactive in that it responds to changes in the environment caused by the incident. Reactive mitigation may reduce the effects of an incident or contain the effects, preventing further escalation.

In its broadest context mitigation also includes such actions as evacuation of workers, since this mitigates the potential harm to the worker(s) by reducing the time during which they are at risk.

3.3 REPRESENTING THE EFFECTS OF MITIGATION

Using the concepts described in this section a schematic representation of the accident scenario can be developed to describe radiological accidents in active handling buildings, as shown in Figure 2.

Using this representation, a plant state can be defined by a vertical line through the graph at the appropriate time. The diagram indicates the spread of activity in time and distance away from the source. The exposure times for the operator and public can be determined clearly. The main feature which cannot be shown using two dimensions is the reduction in the concentration of activity and direct radiation dose rate with distance. The effect of including a number of stages of physical mitigation on the accident scenario is shown by a lightening of the colour, which denotes a reduction in the total activity or radiation passing a given distance from the source. Operator responses which represent mitigation in the time axis are also included.

The key features of accidents resulting from operations in active handling facilities which involve radioactive materials can be seen to be:

- A release of activity from the source, and its duration,
- Gamma radiation from the source,
- Mitigation of the release by residual containment between the source and operating areas of the facility,
- Mitigation of the release to the environment from the operating areas,
- Mitigation of the radiation by residual shielding between source and operating areas of the facility,
- Mitigation of the radiation in the environment by the shielding provided by the building structure,
- Mitigation of the dose uptake by the operator by evacuation or other emergency actions
- Mitigation of the dose uptake to the public by off-site countermeasures.

The number of these which are effective and the extent of the mitigation provided will define the "plant" state for any given accident scenario.

3.4 THE MODEL FOR MITIGATION

Using the idea of mitigation defined above, the incident scenario model described in section 5.1 can be generalised for use in PSA. An accident begins with an initiating event, the effects of which spread outwards in space and in time until some mitigation comes into effect. Depending on whether the mitigation is effective or not, the plant will reach one of several definable plant states. Thereafter the potential accident scenarios will spread out through successive plant states until the physical processes driving the accident are dissipated. If this progression is simplified by assuming that each stage of mitigation will be either fully effective or totally ineffective, then the possible accident scenarios originating from a specific initiating event can be represented as shown in Figure 3.

This is topologically equivalent to an event tree. However it does include the notion of the incident expanding in time as well as space, and therefore reinforces the idea that time itself can play a mitigating role.

4 AEA Technology
4.0 Structure of Reactor Plant PSAs

Comprehensive PSA programmes exist for nuclear reactors therefore the structure of these programmes is examined as an potential basis for the programme.

For nuclear reactors the basic structure of PSA analysis programmes is now well defined\textsuperscript{[4][5][6]}. This structure was formed during the US Nuclear Regulatory Commission (NRC) reactor safety study which was completed in 1975\textsuperscript{[70]}. More recently NRC completed PSAs for four different US nuclear power stations, which were documented in NUREG Report 1150\textsuperscript{[28]}. The NUREG 1150 report effectively summarises the state of the art in nuclear power plant PSA methodology.

Due to the complexity of nuclear reactors and their associated safety systems, nuclear reactor PSA are extremely complex and the PSA methodologies rely on grouping faults together at a number of stages throughout the analysis. The basic elements of the NUREG 1150 risk analysis, and the points where fault sequences are grouped, are shown in Figure-4. These are discussed below.

4.1 LEVEL 1 - INITIAL EVENT-characterisation

Level 1 PSAs normally concentrate on the identification of fault sequences and the derivation of fault trees. This analysis is a major undertaking, with fault trees containing typically 5000 to 10000 gates and basic events, although much larger fault trees have been constructed\textsuperscript{[29]}. Recently, considerable effort has been devoted to assessing the uncertainties associated with fault tree modelling\textsuperscript{[30]} and to the development of methodologies which allow the fault trees to be updated as the plant and operations are modified, i.e. living PSAs.

4.2 LEVEL 2 - ESCALATION WITHIN THE REACTOR CONTAINMENT

Level 2 PSAs concentrate on the accident progression within the reactor building. Most PWR and BWR plants are similar to others in their class, and the way in which accidents can spread within these plants is similar. The progression of the accident is normally modelled using an event tree, which typically contains 100 nodes\textsuperscript{[29]}. The generic event trees are customised for each plant using input data which define key model parameters, such as containment volume and design pressure. Some event trees contain empirical equations which model physical phenomena, such as hydrogen evolution, based on results calculated by earlier nodes in the tree\textsuperscript{[29]}, and are able to use these values to adjust the failure probabilities within the tree. The large size of the containment event tree and the difficulty of presenting such a tree was a major criticism levelled at the NUREG 1150 study\textsuperscript{[29]}

4.3 LEVEL 3 - OFF-SITE CONSEQUENCE CHARACTERISATION

The level 3 PSA aims to determine the probabilistic risk to the public in terms of individual risk and societal impact. This part of the analysis models the dispersion and uptake of many nuclides within the environment for a large number of accident scenarios and presents the results in a form which can convey both the overall risk and the spread of possible risk. The results of the PSA are often presented in terms of cumulative dose, or risk, against frequency, with appropriate percentile ranges, for each risk measure of interest.

4.4 UNCERTAINTY IN REACTOR PSA CALCULATIONS

Following the reactor safety studies much work has been carried out on the evaluation of the effects of uncertainties on the results of the PSAs\textsuperscript{[29][30]}. The complexity of reactor PSAs precludes evaluation of uncertainty for each input data item, and complex statistical techniques are required to create appropriate input data sets and to analyse the results\textsuperscript{[29][30]}

5 AEA Technology
5.0 Current Active Handling Plant PSAs

In general, most active handling plants are not as sophisticated as reactor plants, since the hazard potential of the operations is much smaller. This type of facility also covers a much wider range of plant types and operations than nuclear reactor plants. The range of analyses carried out for active handling plants is as varied as the type of facility, however some key features of these analyses can be described. The concept of three levels of PSA calculation will be continued when describing existing active handling plant PSAs since it allows the analysis to be split up into three well defined stages which have physical meaning in terms of the accident progression.

5.1 LEVEL 1 - INITIATING EVENT CHARACTERISATION

The initiating event sequences for accidents within active handling plants are simpler than the equivalent sequences for reactors. This difference results primarily from the smaller hazard potential of active handling plants, which attract fewer installed safety systems, but often increases the importance of operator intervention.

Event sequences are normally grouped for subsequent analysis. The normal method of analysis is to identify a group or class of similar events, e.g. accidents which result in fuel being dropped within a facility. Each of these groups is then described and analysed separately. Fault trees currently in use generally cover both the initiation and subsequent mitigation phases of the accident scenario. A judgement is often made of the most hazardous source for each group of event sequences and the analysis is then carried out only for that source.

5.2 LEVEL 2 - ESCALATION

Accident progression within active handling plants is generally modelled using a fault tree to determine the frequencies of particular failure modes, e.g. to determine the frequency of a building evacuation or a release to the atmosphere. Event trees are only occasionally used to determine variations in the size of the potential release. The most common use of event trees is to model sequences which lead to different release routes, e.g. where ventilation failures may lead to a release from the building rather than via gas treatment units.

Unlike reactor fault sequences which tend to produce similar events, different fault sequences in active handling buildings can give rise to many different types of hazard. Within a single building it would be common to find some events which could give rise to significant hazards to the operators, but not the public, others which give rise to hazards to the public but not the operators. Therefore there is not the same scope for developing a generic event tree equivalent to the reactor containment event tree.

5.3 LEVEL 3 - CONSEQUENCES OUTSIDE THE BUILDING

The evaluation of the off-site impact of accidental discharges of activity has been the subject of discussions between the Nuclear Site Licensees and the NII. This has resulted in the development of a common methodology for this stage of the assessment\(^{[8]}\), which satisfies the requirements of the NII SAPs. Two main implementation strategies have been adopted to calculate off-site consequences.

One strategy calculates the dispersion effects for each nuclide in the release using the dispersion model equations for each individual situation. The other uses an agreed set of 'average' input parameters for a site and calculates an average dispersion factor for each nuclide. The results are often quoted in Sv/TBq of nuclide. These results are then held in a database and applied to each nuclide in the release during the calculation. The latter provides a much easier method of calculation by hand and provides a greater degree of consistency where many different plants exist on the same site. It does however require the calculation and storage of a large number of nuclide release ratio data sets.

6 AEA Technology
5.4 UNCERTAINTY IN CURRENT PSA CALCULATIONS

Currently uncertainties in PSA calculations are not normally evaluated explicitly. Some methodologies identify items which provide a primary protection function in radiological safety by means of a sensitivity analysis on the fault tree. The effects of uncertainties in other elements of the calculation are not normally evaluated.

6.0 Definition of Level 1 Calculations

The initial stage of the programme analysis is the definition of the source and quantification of the initiating event frequency, see Figure 5. Whereas reactor PSAs aim to calculate the frequency with which the core is uncovered, often called the core melt frequency, the frequency which would be most applicable in active handling operations is the frequency with which a release of activity first meets a barrier which could mitigate its release or which could lead to a loss of shielding or exposure of a worker.

6.1 INITIATING EVENT FREQUENCY

The current version of the WINRISK programme requires that the initiating event frequency is input as a single value, as part of the data describing the accident scenario. This frequency will normally be calculated separately using a proprietary fault tree programme which is capable of carrying out a sensitivity analysis, in line with current practice for manually generated PSAs.

The results of sensitivity analyses on the initiating event fault tree can be entered in the data file to allow the effects of these uncertainties to be evaluated. This process is similar to that used in reactor PSA calculations where the results of the fault tree evaluations are fed to a separate level 2 analysis programme.

6.2 DEFINITION OF THE SOURCE

The source term for an accident must contain a list of the important nuclides and their activities. In general the source used will be the worst case source which is expected to be received in the foreseeable future, and will be defined in terms of readily identifiable operational parameters. For fuel, typical parameters would be initial enrichment, burn-up, mean rating and cooling time.

The calculation of direct radiation dose rates from sources is complex for all but the simplest sources and geometries. Therefore the assessment route requires prior calculation of dose rates at a number of distances from the source. Given this requirement it was decided to use the same file format for storing nuclide information for both the risk assessment and the gamma radiation programme normally used by the author, since this eliminates the need to re-enter or convert source data.

6.3 INITIAL RELEASE PARAMETERS

The source used is normally the total quantity of material which is involved in the accident, since this defines the gamma radiation associated with the source. However, in most accident scenarios only a small fraction of this activity will be released. Release Fractions are currently used to describe the fraction of activity which is released in an event. Release fractions have been assessed for a range of typical scenarios\(^a\), e.g. fuel failure by impact, fuel fires, dropping incidents involving powders or liquids, etc...

Release fractions often vary according to the type of fission product present. In particular noble gases normally have a higher release fraction than nuclides which are locked into the fuel matrix. The release fraction of iodine is often different again because it may be present either as a gas or as a sublimable solid within intact fuel. Thus although a single input for a release fraction is sometimes appropriate, separate release fractions for noble gases, iodine and particulate material provide a more comprehensive description with minimal additional work.
There are cases where additional detail is required to model the release of nuclides from a source. This is achieved by allowing specific release fractions to be specified for any element. Since isotopes of an element are chemically similar, the release fractions for an element apply to all isotopes of that element. The programme allows up to 100 element specific release fractions to be entered.

7.0 Definition of Level 2 Calculations

Level 2 calculations are concerned with modelling the accident progression and definition of the different plant states which might occur. This involves construction of an event tree and manipulation of mitigation factors and other variables which may affect the dose consequence to the workforce or the public. The level 2 calculation is the part where the programme methodology differs most from manual hazard assessment methodologies.

7.1 QUANTIFICATION OF MITIGATION

As indicated earlier, the concept of mitigation is being used to describe a wide range of factors which can affect the hazards to which workers and the public are exposed. Within the programme, mitigation is characterised by the following parameters:

- release fraction data
- data relating to physical barriers
- operator response times
- release durations

7.1.1 Release Fraction Data

Release fractions have been discussed in the previous section as they form part of the definition of the initiating event. It is possible for some events to escalate following failure of some of the mitigation elements in force within the plant, leading to increased releases from the source. Therefore the programme allows up to 5 sets of release fraction (RF) data to be used. Individual sets of RF data can be added or substituted.

7.1.2 Physical Barrier Data

Decontamination Factors (DFs) are used to model the effects of containment and/or filtration on the release of aerial activity, where a large DF represents high integrity containment. The effectiveness of the containment often depends on the form of the nuclides, e.g. normal ventilation filters provide no DF for noble gases. Thus DFs are handled within the programme in the same way as RFs.

Mitigation of gamma radiation can only be handled by an approximate method, since the effectiveness of the shielding provided by different materials depends on the spectrum of photon energies, the geometry of the source and the shielding, the chemical composition of the material and its density. The programme uses attenuation factors to represent the effect of intervening shielding, where attenuation is defined as:

\[
\text{Attenuation} = \frac{\text{Dose rate without shielding}}{\text{Dose rate with shielding}} \tag{I}
\]

Data on the attenuation characteristics of materials can be obtained from reference books, e.g. Ref 35, or by calculations using shielding programmes which can take account of the actual gamma energy spectrum of the fuel, composition of the source and shielding, as well as geometric effects. The use of shielding programmes may also allow the effects of secondary radiation to be taken into account.

The mitigation described by DFs and gamma attenuation may occur before the activity or radiation has reached a worker within the building or between the worker and the public. A switch has been incorporated in the input data to tell the programme if a particular set of mitigation data applies to the worker and the public or to the public alone.

8 AEA Technology

84
7.1.3 Operator Response Times

A number of different evacuation times may be required, to model different operator responses to an accident. For example a short evacuation time would be expected when a local alarm is activated, whereas a longer time would be required to interpret data provided by a display. Failure to evacuate following an event can be modelled using this method by including a limiting evacuation time for some events.

Workers may also evacuate as a result of the initiating event, rather than requiring an alarm to warn them of danger. The programme allows for this by making a distinction between immediate evacuation and prompted evacuation.

7.1.3 Release Durations

Different release durations can be modelled. The release duration can be affected by a number of effects, not least of which is the operational state of ventilation plant.

7.2 FAILURE PROBABILITY DATA

Failure data is associated with each level of mitigation used to model the event. The failure data can be entered as a probability or as a failure rate (λ) and either a repair time or test interval (t). If the latter option is used, the failure probability (PFa) is calculated using the equation:

\[ P_{\text{fa}} = 1.0 - \frac{1.0 - e^{-\lambda t}}{\lambda t} \]  

This equation assumes a constant failure rate for equipment. Whilst the failure characteristics of all the equipment will not conform to a constant failure rate model this approximation is commonly used in PSA applications and is justifiable where plant specific data is collected and updated regularly.

7.3 EVENT TREE CONSTRUCTION

The event tree used to model the progression of the accident is defined by a set of data items for each node. The programme constructs the event tree from the individual node data sets, collecting the appropriate mitigation parameters for each end point in the process.

7.3.1 Event Tree Data

The first node contains data relating to the initiating event. This consists of

- A short description of the event
- The initiating event frequency
- An equation which defines which release fraction data sets are to be used to define the initial release of activity.

This equation can also define other data items which apply to all plant states, such as the area of the plant where the accident may occur or a release duration.

The programme is currently able to handle up to 10 stages of mitigation, or event tree nodes. Each node is described by the following information:

- A short description of the mitigation stage
- The failure probability or failure frequency and repair time/test interval
- An equation which defines which mitigation data sets are to be used to define the effect of the accident when successfully mitigated.
- An equation which defines which mitigation data sets are to be used to define the effect on the accident when the mitigation fails.

Thus the event tree may contain up to 2^6 end points for an accident scenario. In practice most accident scenarios can be described using around 6 event tree nodes, which gives rise to a more manageable 64 end points.

9 AEA Technology
7.3.2 Event Tree Construction

The event tree section of the model calculates three main sets of parameters: the failure frequency at each node of the event tree, the total mitigation provided to the operator at each node and the total additional mitigation provided to the public at each node. Total mitigation is calculated separately for airborne activity and gamma attenuation. The total mitigation for each end point of the event tree is passed to the level 3 analyses for the evaluation of dose uptake.

To optimise the conflicting requirements of minimising both memory and calculation time the programme initially calculates and stores all the node frequencies and cumulative mitigation data for each node which forms part of the 'top line' of the event tree. Thereafter only the parts of the tree which have changed are updated when the next end point is calculated. Since the data is calculated one line at a time, the event tree is presented in a different form to a standard tree. The event tree printed by the programme indicates a success by a horizontal line to the next node and a failure by a vertical line to the next node, as shown in Figure 6.

Total mitigation of an accident as the result of a stage of mitigation will result in all sub-branches of that node having no consequence. Therefore the programme truncates the calculation for any further nodes on that line of the tree, and carries the node frequency to the end point column. The calculation is then continued at either the failure node of the current end point or the next success node of the preceding event. This effect is shown in Figure 7.

8.0 Level 3 Calculations for the Public

The dose uptake model for the public consists of two separate models, one for direct radiation and one for all the pathways associated with the release of airborne activity. These are discussed separately below.

8.1 AERIAL DISPERSION MODEL

The aerial dispersion model used to evaluate the transport of radioactive material from the incident building is based on that recommended by the National Radiological Protection Board (NRPB). The model uses a Gaussian plume model, which includes terms to account for the effects of weather conditions, wind velocity profiles, release height and release duration. The effects of removal processes on the air and ground concentration of radionuclides released have been examined in References 26 and 27. These have been summarised in Reference 28. The model described in Reference 28 has been accepted by the NII as being appropriate for use by licensees of UK Nuclear Licensed Sites.

Releases may be from an isolated point or stack where the effects of buildings are negligible or from a simple building or group of buildings. Releases from a complex group of buildings are treated as isolated ground level releases as this is the most conservative calculation. In practice, most active handling buildings are located on a site which includes many other buildings which may be located relatively close to each other. Thus, initially, only the model associated with isolated stack releases have been included, since this includes the ground level release model.

Four types of dose uptake by the public have been identified for use in hazard assessments within AEA Technology: categorisation dose, ERL dose, individual risk dose and PSA dose. These are defined below.

The categorisation dose is an estimate of the maximum possible dose to the public. It is used to determine the level of scrutiny to which a modification to a plant will be subjected. The calculation has the following features:

- The weather conditions used are those which give the maximum dose to the public.
- The exposed person is assumed to be an adult member of the public.
- The exposed person is assumed to be in the open on the centre line of the plume at the point of maximum off-site dose.
- Exposure pathways of cloudshine and inhalation are included in the assessment.
- The effects of deposition by rain are excluded.
The effects of re-suspension of deposited activity are excluded

The ERL dose is calculated for comparison with Emergency Reference Levels which are used to identify the requirement for, and scope of, off-site countermeasures. The dose calculation is for an acute intake of nuclides and is very similar to the categorisation dose. The differences from the categorisation dose are:
- Ground shine integrated over two weeks is included.
- The exposed person is a member of the age group which will give the maximum dose.

For the purposes of these assessments the public is considered to fall into groups which are represented by adults, children aged 10 years and 1 year old infants. The definition of these reference groups was carried out by the International Commission on Radiological Protection (ICRP).

The individual risk dose is that used to calculate the average individual risk to a member of the public. The main features are:
- Calculations are combined for all weather conditions based on their probability of occurrence.
- Dispersion is assumed to be into a 360° sector. This is equivalent to assuming that there is an equal probability of the wind blowing in any direction, and that the public are evenly distributed around the plant.
- The exposed person is a member of the age group which will give the maximum dose.
- The individual is located at the worst plausible location. This is the location at which people may reasonably be expected to be, which gives the highest dose. Typical examples would be habitation or a nearby frequently used road. If there is no plausible location within 1 km then the dose should be calculated at 1 km.
- The effects of rainfall should be included.
- All pathways should be included. Dose uptake by ingestion is limited to the public legal dose limit of 5 mSv over time, as countermeasures are required to be enforced to ensure that this limit will not be breached in an individual’s life time. In practice countermeasures are likely to be enforced at the lower EC interdiction levels.
- In general countermeasures should not be included in aerial dose uptake assessments, except for very large releases, to ensure reasonable conservatism.
- Re-suspension of deposited activity is included.
- Integration times of 50 years are used for re-suspension, ground shine and transfer factors associated with ingestion doses.
- Allowance is made for people being inside buildings during an accident, when calculating groundshine and cloudshine.

The PSA dose has been defined to meet the requirements of NII Safety Assessment Principle 42. The dose calculation is very similar to that given above except that the plume is assumed to disperse into a 30° sector and the member of the public is always assumed to be present in that sector.

A summary of the calculation parameters used is given in Table 1.

8.2 DIRECT RADIATION MODEL (Public)

The dose rate from the source must be provided as part of the input data. The input dose rate can be provided for a number of distances from the source. This range should ideally cover the full range where a point source approximation would not be valid. The dose rate to the public is determined from last dose rate point given, using the inverse square law. The exposure time will be determined by the nature and speed of off-site countermeasures. For assessment of categorisation dose there is a recommended default value of 1 hour for exposure to direct radiation.

9.0 Level 3 Calculations for Workers

The level 3 calculation for dose uptake by the workers is focused on the potential dose uptake by the worker involved in the incident. The main exposure routes for a worker are inhalation of airborne activity and direct radiation. These are discussed separately below. Dose uptake calculations for the worker are carried
out for a minimum separation distance between the worker and the event, for categorisation, and for the most likely separation distance, for risk assessment.

9.2 INHALATION DOSE TO WORKERS

The inhalation dose obtained by integrating the concentration of activity over the exposure time for the worker. The equations used to calculate the integrated air concentration is based on work recently carried out within AEA Technology, which derives equations for instantaneous ‘puff’ releases and for continuous releases. These have been developed to include a release which covers a finite but significant fraction of the exposure time.

Both models are based on the concept of an effective dilution volume which expands with time due to random mixing processes. The effectiveness of the mixing processes are modelled using a parameter which describes the degree of turbulence in the affected area. Since most active handling buildings are ventilated by forced ventilation systems the value of α will be dominated by the air change rate of the area rather than molecular diffusion or natural convection effects.

The ‘puff’ model provides a good model of short term releases. It assumes that the activity is uniformly distributed within the dilution volume at all times. This can then be integrated to give a simple equation for the dose to the worker.

The continuous release model is derived from the puff model described above. In the continuous model, the contribution of the activity released in each, short, time interval is modelled as if it were a separate puff of activity. This can then be integrated twice to give a simple equation for the dose to the worker.

The above two models apply to very short and very long release times, however for cases where the worker exposure time is slightly greater than the release time, there is a significant over estimate of the dose uptake to the worker if the puff model is used. In addition the change from one model to the other creates a discontinuity is the predicted dose uptake. Therefore an impulse release model has been derived.

In the impulse model, the concentration at a point is obtained by integrating the continuous release model over the total release time. The concentration is then obtained by integrating the concentration over two separate time intervals; one representing the release period and one representing the period of time between the end of the release and the evacuation of the operator, to obtain the integrated air concentration and hence dose to the operator.

9.2 MODELLING THE DURATION OF EXPOSURE

All dose uptake models require specification of the time at which the cloud first reaches the worker and the time at which the worker evacuates. The first of these can be calculated directly from the equations governing the dilution volume. The second may depend on the activation time for the alarm system or may be the direct result of an obvious event.

If the warning is provided by the event or by the activation of a gamma alarm the exposure time will be equal to the evacuation time specified in the input data. If the operator response is triggered by an activity in air monitor, the allowance is made for the cloud to reach the monitor and for the integrated air concentration to exceed the alarm level.

9.3 DIRECT RADIATION DOSE FOR WORKERS

The dose rate from the source must be provided as part of the input data. The input dose rate can be provided for a number of distances from the source, which should cover the full range where the point source approximation is not valid. For cases where the worker is at a distance greater than the last source data point, the equation used to calculate the dose rate is the same as that used for the public. Where the worker is at a location which is within the range of axial locations for which dose rate data is provided, the dose rate is obtained by linear interpolation between adjacent data points. Dose rates at gamma monitors located in operating areas are calculated in the same manner to those for the worker.
Calculation of Risk Parameters

For each end point, the results of the frequency and dose calculations are combined to determine the key risk measures discussed below.

10.1 DOSE/FREQUENCY STAIRCASE

The dose contributions for the public are calculated by adding the PSA dispersion model and direct radiation doses. The end point frequency is then added to the running total for that dose uptake band. The SAPs require the frequencies of occurrence to be calculated for the dose bands given in Figure 1; > 1 Sv, 100 mSv - 1 Sv, 10 - 100 mSv, 1 - 10 mSv and 100 μSv - 1 mSv. To give a better picture of the risk profile presented by the accident scenario, The WINRISK programme also calculates frequencies for the following dose bands: 10 μSv - 100 μSv, 1 μSv - 10 μSv and 0.1 μSv - 1 μSv.

Although not required by the SAPs, the event frequencies in each dose band are also calculated for workers. The total dose to the worker is calculated using the PSA inhalation dose and direct radiation dose.

10.2 INDIVIDUAL RISK

The individual risk dose uptake calculation takes into account the fact that the plume may be blowing away from the individual, by modelling a plume dispersion into a 360° sector. Whilst this method assumes that the probability that the wind blows in a given direction is equal for all directions, it also assumes that the member of the public is located at the same distance from the plant in all directions. The overall effect of these two simplifying assumptions is pessimistic in most cases.

To calculate the individual risk to a member of the public or the workforce, a risk factor must be applied to model the conditional risk of a fatal cancer. Two values of risk factor are recommended by the NRPB75 for both the public and the workforce, one for low dose and low dose rates and a higher value where the total dose is greater than 100 mGy and the dose rate is greater than 0.1 mGy/min.

The estimated annual individual risk is calculated from the product of the total dose, the event frequency and the appropriate risk factor. The total risk is obtained by adding the individual contributions from each end point.

10.3 FREQUENCY OF OFF-SITE COUNTERMEASURES

The frequency with which off-site countermeasures may be required is calculated using the ERL dose calculation and the maximum direct radiation dose to the public. If the total of these two doses is greater than the lower emergency reference level then the calculated frequency is added to the appropriate frequency accumulator.

10.4 CATEGORISATION DOSE

The maximum categorisation dose to the public and the workers is updated every time an end point is calculated. The end point number of the event which gives rise to the maximum dose to each of the critical groups is printed out at the end of the calculation.

10.5 SAFETY CASE SUMMARY FILES

As a first step towards providing an integrated, or living, PSA package the WINRISK programme can create or update a file which contains a summary of individual risk assessments. This can be used to monitor the effect of changing a single accident scenario on the risk from an individual building. Where an existing file is updated then the new risk profile can be added to a set of profiles for other accident scenarios or can update an existing accident scenario profile.

13 AEA Technology
The dose/frequency staircase results and individual risks for both the public and the workforce are stored in the file, along with a description of the accident scenario and an accident scenario identifier. The total frequency for each dose/frequency staircase range, and total individual risk, across all the accident scenarios are calculated and stored. A description of the group of accident scenarios held within the file can be also be added.

11.0 Sensitivity Analysis

Two forms of sensitivity analysis can be carried out; one varies the individual data items by an order of magnitude, the other investigates the effects of each mitigating barrier being permanently in a failed state.

The sensitivity analyses are carried out to quantify the sensitivity of the following risk assessment parameters to the input data;

- Dose / frequency staircase for public and workers
- Individual risk for public and workers

11.1 ORDER OF MAGNITUDE SENSITIVITY ANALYSIS

The sensitivity analysis is carried out for the following input data items, with each item being incrementally individually, in turn;

- Release Fractions
- Decontamination Factors
- Gamma attenuations
- Release duration
- Operator exposure times
- Barrier failure frequency or failure probability
- Each sensitivity result input for the initiating event frequency

This form of sensitivity analysis is rudimentary in that the variation of each input parameter is not related to the uncertainty in its value, and it assumes that there are no significant interaction effects between different parameters. It does however indicate which of the parameters are most significant, in sufficient detail to allow a subsequent manual assessment of the importance of each result. This form of analysis also places minimal requirements on the quality of input data.

11.2 SEARCH FOR SAFETY MECHANISMS

The second sensitivity analysis sets the failure probability for each mitigation stage to 1 in turn. This analysis is used to identify those items which if "removed or in a permanently failed state" would lead to a risk which would breach one of the risk criteria. This class of safety systems are called "Safety Mechanisms" and are the subject of special requirements laid down in each Nuclear Site License.

11.3 PRESENTATION OF SENSITIVITY ANALYSIS RESULTS

The sensitivity analysis compares a set of the recalculated results with the main requirements of the NII SAPS. The parameters which are compared are the dose/ frequency staircase results and the risk to the workers. Results which exceed 10% of the BSL or the BSO are identified on the output.

The results of the sensitivity analysis are then ranked in order of descending risk to the public and, separately, in descending order of risk to the workers. These are printed out to assist in the identification of those factors to which the risk is most sensitive.
12.0 Conclusions

This paper has described a computer programme, WINRISK, which has been written to carry out radiological risk assessments for active handling plants, which satisfy the requirements of the NII Safety Assessment Principles and AEA Technology's Corporate Safety Instructions.

Risk assessment programmes for nuclear reactor plant has been reviewed and compared with risk assessment methods for active handling plants. This comparison has highlighted differences in scale and techniques used. The WINRISK programme uses a combination of these two approaches.

The main features of the WINRISK programme have been described in terms of the three levels of accident progression used in reactor PSAs. An outline of the calculational methods used has been given. The sensitivity analysis used within the programme covers both failure data and data relating to the mitigating effects of key plant items.

The programme can create or update files which hold summary data for a group of accident scenarios. This allows a summary of all accident scenarios associated with a facility to be held together in one file.
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92   AEA Technology
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Fig 1 Dose - Frequency Staircase

Fig 2 Event Progression Showing Mitigation

Fig 3 Generic Event Scheme
Fig 4 Elements of NUREG 1150 Risk Analysis

Fig 5 Elements of WINRISK Risk Assessment

Fig 6 Event Tree Representation Within WINRISK
Fig 7  Representation of Total Mitigation Within WINRISK
SAFETY REQUIREMENTS

FOR NUCLEAR FUEL CYCLE FACILITIES IN GERMANY

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Abstract

The general requirements for licensing of nuclear facilities in Germany are laid down in the Atomic Energy Act (AtG) and the subordinate Radiation Protection Ordinance (StrlSchV). Below the legal requirements technical safety criteria for nuclear fuel cycle facilities have been established. These criteria are the framework for the safety assessment in the licensing procedure for construction, operating and accident analysis. In this paper the German safety requirements for fuel fabrication and enrichment facilities are presented. Basic safety aspects as well as special requirements and recommendations are discussed. Examples on safety related design and construction of fuel fabrication facilities are presented with emphasis on MOX fuel fabrication.
1 Framework of Regulations

1.1 Overview

The framework of regulations on licensing and safe operation of nuclear fuel cycle facilities is shown in figure 1. The general requirement are laid down in the Atomic Energy Act of 1976. The so-called Radiation Protection Ordinance (amended edition of 1989) is a subordinate decree, where for instance the dose limits are determined.

The Safety Criteria and Recommendations were established for the design and the operation of front-end fuel cycle plants. The Safety Criteria were published by the Federal Minister of Interior within the Handbook of Reactor Safety and Radiation Protection in 1983.

The basic level of the framework is covered by technical standards and regulations for example the DIN standards and KTA regulations.

Figure 1: Framework of licensing regulations in Germany
1.2 Atomic Energy Act

The principles of licensing of nuclear facilities are formulated in paragraph 7. According to this, fuel fabrication facilities have to be licensed in the same way as nuclear power plants. The essential requirements are:

- Same licensing procedure (application for licensing, safety report and short description of the plant for public information),
- safety analysis and examination by independent expert organizations,
- a public hearing has to be carried out; this is also required for significant modifications of an existing facility.

Licensing authority is the state government, local authorities may contribute, supervising authority is the Federal Minister of Environment, Nature Protection and Reactor Safety.

Further responsibilities which have to be taken into account are:

- Liability insurance on nuclear accidents, for fuel fabrication plants up to 200 Mill. DM maximum.
- Safeguards by IAEA and EURATOM.
- Physical protection against unauthorized access to the nuclear material.

1.3 Radiation Protection Ordinance

The radiation protection is based on ICRP Principles. The basic requirements are in detail formulated in the "Ordinance about Protection against Injuries by Ionizing Radiation".

Protection of employees as well as protection of population and environment have to be realized in a sufficient manner.
Limitations of individual doses for population are given for normal operation including deviations (para 44) and for design basis accidents (para 28.3).

For the calculation of dose values for population the doses by direct radiation, inhalation and intake of radionuclides with food have to be taken into account. For design basis accidents the accumulated dose values up to 50 years must be calculated. Special guidelines how to perform these calculations are issued.

For workers in the plant the given individual dose limitations are based on external radiation and incorporation.

The main annual radiation dose limits for normal operation for workers and population are shown in table 1. The derived German annual limits for intake of U, Pu and Am are given in table 2. The limits are not additive.

Additionally to these limits the 'ALARA' principle ('as low as reasonably achievable') has to be attended for design as well as in operation.
Table 1: Main German Annual Radiation Dose Limits

<table>
<thead>
<tr>
<th></th>
<th>Employees 18 or over</th>
<th>Population</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Direct Radiation</td>
<td>Aerial Discharges</td>
</tr>
<tr>
<td>Whole body effective dose</td>
<td>50</td>
<td>1.5</td>
</tr>
<tr>
<td>Individual organs and tissues, lens of the eye</td>
<td>150</td>
<td>0.9</td>
</tr>
<tr>
<td>Thyroid</td>
<td>150</td>
<td>0.9</td>
</tr>
<tr>
<td>Bone Surface, Skin</td>
<td>300</td>
<td>1.8</td>
</tr>
<tr>
<td>Hands, Arms, Feet</td>
<td>500</td>
<td></td>
</tr>
</tbody>
</table>

Notes:

(1) For employees a total whole body lifetime doses of 400 mSv has not to be exceeded.

(2) The limit of whole body effective dose for direct radiation includes contribution from discharges.
Table 2: Derived German annual limits for U, Pu and Am intake

<table>
<thead>
<tr>
<th></th>
<th>Annual Intake Limit, Bq</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Inhalation</td>
<td>Ingestion</td>
</tr>
<tr>
<td>U all forms</td>
<td>500</td>
<td>300 000</td>
</tr>
<tr>
<td>U dioxide</td>
<td>500</td>
<td>3 000 000</td>
</tr>
<tr>
<td>UF₆</td>
<td>30 000</td>
<td>300 000</td>
</tr>
<tr>
<td>Pu all forms</td>
<td>100</td>
<td>20 000</td>
</tr>
<tr>
<td>Pu nitrate</td>
<td>100</td>
<td>200 000</td>
</tr>
<tr>
<td>Pu dioxide</td>
<td>400</td>
<td>2 000 000</td>
</tr>
<tr>
<td>Am-241</td>
<td>100</td>
<td>20 000</td>
</tr>
</tbody>
</table>
Safety Criteria

2.1 Status of the Safety Criteria

The 'Safety Criteria for Uranium Enrichment Plants and for Fuel Fabrication Facilities' have originally been written by GRS on behalf of the Federal Ministry of Interior. After discussions in different advisory groups they were officially issued by the Federal Ministry in 1983.

The Safety Criteria are essentially a more detailed determination of the safety related 'state of science and technology' as it is demanded in general by the Atomic Energy Act and the Radiation Protection Ordinance.

The Safety Criteria were used in the licensing procedures of German fuel fabrication plants, namely the former RBU, ALKEM and NUKEM at Hanau and the EXXON facility at Lingen (Lower Saxony), which are (except NUKEM) now all owned by Siemens AG.

A revised and expanded version of the safety criteria, issued as GRS-publications /1a - c /, was approved by the Reactor Safety Commission in 1991. They are not yet officially issued by the Federal Ministry, but used as guidelines in ongoing licensing procedures.

2.2 Main Topics

The main topics of the Safety Criteria are:

- Siting
- External Events
- Fire and Explosion Protection
- Chemical Hazard, Corrosion
- Protection against Leakage
• Resistance against Overpressure

• Radiation Protection

• Criticality Safety

• Construction and Design

• Quality Assurance

• Plant Operation

• Accident Analysis

• Waste Management

• Emergency Measures

• Decommissioning

Some of these topics will be discussed in more detail in the following sections.

2.3 Fire and Explosion Protection

Prevention of uncontrolled fire and explosion is an important safety goal especially in Mox fuel fabrication, but also in UO₂ fuel fabrication and enrichment. To ensure that this goal will be reached, different types of safety measures are required.

Criteria for design and construction:

• Separation of fabrication and storage sections by fireproof walls.

• As far as possible use of fire resistant materials for construction of container and glove boxes.

• Separation of safety systems, ventilation equipment, off-gas filters

• Separation of supply systems for hydrogen and other burnable media

• Protection of ventilation systems.
Criteria for operation:

- Avoidance of ignition sources and if possible inflammable materials (for instance gas mixtures, surveillance of radiolysis gas production).

- Operation of fire detection systems.

Criteria for fire fighting:

- Stationary and mobile fire fighting systems

- Automatic fire extinguishing systems for areas with dispersible Pu and Pu storage or with difficult access

- Fire fighting by water only outside of areas with criticality risk

For example some topics of the actual design of the new Siemens MOX fuel fabrication facility are listed [2]:

- Strict partitioning of the plant into fire zones, automatic separation of connections (pipes, ventilation system) in case of fire

- automatic operation of extinguishing systems (sprinkler, gas), started by fire detectors

- fire detectors in glove boxes

- an own specially equipped fire brigade.

2.4 Radiation Protection

As discussed in section 1 the basic requirements and dose limitations are laid down in the Radiation Protection Ordinance. In the plant specific Safety Criteria the more detailed requirements and measures to realize the radiation protection are determined.
These are essentially:

*The Barrier Concept*

The Barrier Concept is important for MOX fuel fabrication. Always two independent barriers to the environment and at least one barrier to protect the workers are required. If glove boxes and working rooms are part of the barrier concept, a gradient of pressure between the barriers is to be maintained. Figure 2 shows a sketch of the barrier concept as it is realized in the new Siemens MOX Fuel Fabrication Facility /2/.

![Diagram of the Barrier Concept](image)

\[ \Delta P_3 = 250 \text{ pa} \quad \Delta P_2 = 75 \text{ pa} \quad \Delta P_1 = 20 \text{ pa} \]

Figure 2: Barrier Concept for the Mixed Oxide Processing Plant with Graduated Decreases in Pressure /2/

In Table 3 the derived requirements for the design and layout for ventilation of the Siemens MOX fuel fabrication plant are given.
### Table 3: Layout of Ventilation of the Siemens MOX fuel fabrication plant

<table>
<thead>
<tr>
<th>VENTILATION</th>
</tr>
</thead>
<tbody>
<tr>
<td>Off-Gas Filtration</td>
</tr>
<tr>
<td>Glove boxes</td>
</tr>
<tr>
<td>Room Ventilation</td>
</tr>
<tr>
<td>Uranium handling</td>
</tr>
</tbody>
</table>

Gradient of pressure to contain radioactivity
(250 pa from glove boxes to outside atmosphere)

Protection of filters from corrosive chemicals (off-gas scrubbers) and from fire and fire products (soot and pressure buildup)

---

**Radiation protection inside the plant**

Measures against contamination, incorporation and external radiation have to be carried out. For components, which contain Plutonium radiation shielding devices (gamma and neutron) are demanded. For installations, where large amounts of Plutonium are processed in addition removed handling and/or automatic control is provided.

**Radiation Protection for population and environment**

For the proposed values of discharges of radionuclides with off-gas and waste water under normal operation conditions it has to be proved that the dose limits (of the Radiation Protection Ordinance) will not be exceeded for people living near the plant site, even under unfavorable assumptions.
For Pu manufacturing additionally activity control measurements outside of the plant are provided.

As an example the design topics related to radiation protection for the new Siemens MOX facility are given /3/.

Radiation Protection in the new Siemens MOX Plant

- Limit for effective equivalent dose 10 mSv/a for workers
- Design Principles:
  - Minimization of inventory in working area,
  - shielding within glove boxes, use of double walled glove boxes,
  - use of combined n/γ-shieldings (polyethylene + lead + neutron absorber),
  - minimization of manipulation by staff by means of automated processes,
  - application of the barrier concept to prevent contamination and incorporation,
  - air-, surface- and personal monitoring.
- All working places were calculated with the special computer program PUDOL
- Results (with annual throughput 120 t HM/a, mean Pu-content 7.2 %):
  - < 10 mSv/a will be well observed
  - collective dose < 1 Sv/a

2.5 Criticality Safety

Criticality safety in handling, manufacturing and storage of fissile material is based on the double failure principle, which is in detail written in the German standard DIN 25 403.

In general the proof of criticality safety is based on calculations. For systems with simple geometric shape (cylinder, slab or sphere) data from handbooks may be use, for instance the German criticality safety handbook (ed. GRS). Suitable safety margins
must be taken into account. In table 4 the safety factors, which shall be applied to critical limits are given.

In more complicated cases like arrays of fissile material or special geometric shapes, criticality safety has to be proved by calculation. Generally a value of $k_{\text{eff}} < 0.95$ is required including uncertainties.

For normal operation those conditions, which lead to the highest k-value have to be considered, such as the highest density of fissile material, moderation, chemical compound etc. For example in the criticality analysis of MOX fuel fabrication a residual humidity of 3 wt % of water was assumed for dry MOX.

Neutronic interaction of adjacent components must be checked, interspersed moderation effects are to be taken into account.

Typical situations, which are recommended to be analyzed:

- Transportation of containers with fissionable material, especially Pu,
- accumulation of fissile material in washing columns or filters,
- leakage of fissile solution, possibly into an adjacent component,
- transfer of solution to a container of different size,
- transfer of fissile material between areas with different safety concepts.

To realize criticality safety, essential safety concepts, which are described in detail in the DIN 25 403 standard are recommended:

- Safe geometry, limitation of dimensions,
- mass limitation, double batching has to be considered,
- moderation control, to be applied only for handling of dry material,
- limitation of enrichment, control by at least two independent ways is required.
- If neutron poisoning is used as part of a safety concept, additional measures for quality assurance are required, to ensure the efficacy of the neutron absorber.
Priority should be given to technical safety measures, such as limitation of dimensions instead of administrative ones.

A criticality alarm system is required for all areas, were amounts of fissile materials are handled or stored, which are large enough to reach criticality under certain conditions. Instructions for the alarm case have to be provided.

For the new Siemens MOX fuel fabrication plant criticality safety is achieved by safe geometry, neutron poisoning and moderation control. For non-standard handling the safety concept of mass limitation is applied. For criticality analysis a Pu vector of 95 % Pu-239, 5 % Pu-240 was assumed.

Table 4: Recommended safety factors for the limitation of critical parameter

<table>
<thead>
<tr>
<th>Limited Parameter</th>
<th>Safety factor to be applied on the critical parameter</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass</td>
<td>0.45</td>
</tr>
<tr>
<td>double batching excluded by design</td>
<td>0.80</td>
</tr>
<tr>
<td>volume of sphere &lt; 5 l</td>
<td>0.80</td>
</tr>
<tr>
<td>volume of sphere &gt; 5 l</td>
<td>0.75</td>
</tr>
<tr>
<td>Diameter of cylinder &lt; 50 cm</td>
<td>0.90</td>
</tr>
<tr>
<td>Diameter of cylinder &gt; 50 cm</td>
<td>0.85</td>
</tr>
<tr>
<td>Slab thickness</td>
<td>0.80 - 0.90</td>
</tr>
<tr>
<td>Concentration</td>
<td>0.50</td>
</tr>
</tbody>
</table>
2.6 Accident Analysis:

In addition to the required measures to ensure safe operation, a systematic screening of design and operation is demanded to identify possible accidents. Existing experience shall be considered. This leads to the derivation of design basis accident scenarios. The methodology for analysis is primarily deterministic. Additionally fault tree analysis or probabilistic methods are recommended. The basic types of DBA, which have to be considered are listed in the Safety Criteria. The design basis accident scenarios for MOX fuel fabrication are listed in table 5.

There are two acceptable possibilities of treatment:

- Prevention by design,
  - radiological consequences are less than dose limits.

The dose limits of DBA (only for population) are determined in the Radiation Protection Ordinance (§ 28.3):

- 50 mSv effective dose, integrated over 50 years.

Additionally protection against external events is required:

- For natural events such as storm, snow, low temperature, flooding, lightning and earthquake protection by proper siting or by design is required.

- For man-caused events, such as toxic substances, airplane crash, external pressure wave the evaluation of adequate risk reduction is provided.

In the Siemens MOX fuel fabrication plant protection by design against airplane crash and criticality by fast neutrons is realized for Pu storage and MOX processing areas.
<table>
<thead>
<tr>
<th>Event</th>
<th>Systems</th>
</tr>
</thead>
<tbody>
<tr>
<td>Explosion</td>
<td>Calcination or sintering furnace, evaporator, accumulation of ammonium nitrate, hydrogen accumulation (during storage of plutonium nitrate), spontaneous failure of pressure vessels or supply systems for gas or chemicals on site</td>
</tr>
<tr>
<td>Local Fire</td>
<td>Extraction or ion exchange unit, contaminated waste or filter material, solvents</td>
</tr>
<tr>
<td>Criticality</td>
<td>Plutonium solution, moderation of undermoderated plutonium powder.</td>
</tr>
<tr>
<td>Leakage</td>
<td>Pu-solutions or -powders</td>
</tr>
<tr>
<td>Mechanical Impact</td>
<td>Drop of heavy loads, damage to glove boxes or gloves</td>
</tr>
<tr>
<td>Loss of Supply Systems</td>
<td>Electric power, gas, water, pressurized air, off-gas-and ventilation systems, cooling systems, impact from released chemicals</td>
</tr>
<tr>
<td>Earthquake</td>
<td>All facilities on site</td>
</tr>
</tbody>
</table>
Finally some results of accident analysis for the new Siemens MOX fuel fabrication facility are given /4/:

<table>
<thead>
<tr>
<th>Event</th>
<th>Calculated radiological impact of DBA with possible Pu-release (effective dose equivalent, mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Local fire in HEPA filters</td>
<td>$2.7 \times 10^4$</td>
</tr>
<tr>
<td>Criticality ($5 \times 10^{19}$ fissions assumed)</td>
<td>8.3 (from iodine)</td>
</tr>
<tr>
<td>Leakage (powder resp. solution)</td>
<td>$2.2 \times 10^{-7}$</td>
</tr>
<tr>
<td>Crash of glove box</td>
<td>$3.7 \times 10^{-12}$</td>
</tr>
<tr>
<td>Drop of heavy loads (container, Fuel elements, fuel rod)</td>
<td>$3.6 \times 10^4$</td>
</tr>
<tr>
<td>Earthquake</td>
<td>$1.2 \times 10^{-4}$</td>
</tr>
</tbody>
</table>

For earthquake the basis for calculations was a strength of 6 - 7 on the MSK-scale with horizontal accelerations of 2 - 2.45 m/sec².

The design against airplane crash is realized by 1.8 - 2.0 m steel-concrete and a two shell structure for the process building. As this event is not a DBA, the possible radiological impact must not be calculated.
References

/1a/ Sicherheitsanforderungen für Urananreicherungsanlagen nach dem Gasultrazentrifugenprinzip, GRS-Ausarbeitung
GRS-A-1869, Dez. 91

/1b/ Sicherheitsanforderungen für die Herstellung von Leichtwasser-Reaktor-
Brennelementen mit niedrig angereichertem Uran, GRS-Ausarbeitung
GRS-A-1870, Dez. 91

/1c/ Sicherheitsanforderungen für die Herstellung von Uran/Plutonium
Mischoxid-Brennelementen, GRS-Ausarbeitung
GRS-A-1872, Dez. 91

/2/ G. Brährler - Present Status of the Hanau MOX Fuel Fabrication Facility
with Emphasis on Safety Issues, Proc. of the CSNI Symposium on the
Safety of the Nuclear Fuel Cycle
Brussels, 3rd - 4th June 93

/3/ Technical Concept of the New Siemens MOX-Fabrication Plant,
Anglo/German Co-operation in the Nuclear Fuel Cycle Field,
3rd Meeting, 24 - 26 May 94, Hanau

/4/ M. Schmid, Siemens AG - Power Generation Group (KWU)
private communication, August 94
The Atomic Energy Act of 1954, as amended, requires the regulation of nuclear materials to provide for the common defense and security and to protect the health and safety of the public. The Act authorizes the issuance of licenses to applicants "... who are equipped to observe and who agree to observe such safety standards to protect health and minimize danger to life or property as the Commission may by rule establish; ...."

Clearly, on the basis of the Act, the Nuclear Regulatory Commission's regulatory role is to ensure that licensees possessing nuclear material do not use that material in any way that imposes undue risk to health, life, or the environment. This means that a licensee shall identify any and all aspects of its operations that could exceed regulatory limits as a result of the release of radioactive material or the exposure of individuals to radiation. This also means that the NRC shall approve protective measures intended to prevent such releases or exposures. To fulfill the NRC's regulatory role, the Office of Nuclear Material Safety and Safeguards, Division of Fuel Cycle Safety and Safeguards, has several responsibilities:

1. establishing the regulatory requirements and guidance for adequate safety and safeguards;
2. granting a materials license based on the applicant's ability to provide adequate safety and safeguards according to regulatory requirements; and
3. inspecting, monitoring, and assessing the licensee's activities to ensure that the licensee is providing adequate safety and safeguards in compliance with regulatory requirements and guidance.

The NRC's first step in meeting its regulatory responsibilities is to define the necessary requirements and guidance for protecting the public health and safety and safeguarding nuclear material. To this end, the NRC is in the process of rewriting its Nuclear Material Licensing Regulation, 10 CFR Part 70. The impetus for this rewrite is to require licensees to systematically identify and treat site unique hazards in an integrated manner, address outlier event sequences that are precursors to unacceptable events, reduce uncertainties about event sequences, and document the safety and safeguards rationale supporting the licensing basis for concluding adequate safety and safeguards.
2.0 Integrated Safety Analysis

One of the major concepts that emerged as a result of the development of the proposed new Part 70 regulation is the Integrated Safety Analysis (ISA). An ISA is a systematic examination of a facility's processes, equipment, structures, and personnel activities to ensure that all relevant hazards that could result in unacceptable consequences are adequately evaluated and appropriate protective measures are identified. The basic hazards to be treated are nuclear criticality, process releases of radioactive material, chemical hazards, and fire hazards. The ISA is expected to form the basis of a safety program that requires adequate controls and systems to be in place to ensure the safe operation of the facility. In general, the ISA should:

- describe the hazardous materials, site, structures, equipment, and process activities at the fuel cycle facility;
- identify and provide a systematic analysis of hazards at the facility;
- identify potential event sequences, especially those that would result in unacceptable consequences;
- identify and describe the controls (i.e., site, structures, systems, equipment, human actions, or components) that are relied upon to prevent unacceptable events or mitigate their consequences; and
- examine the external events resulting from meteorological and seismological phenomena and their potential for causing unacceptable events at the facility. Meteorological phenomena would include tornados, hurricanes, maximum precipitation, and flooding.

The concept of the ISA emerged as a result of a study, NUREG-1324, which concluded that the regulatory basis could be defined and overall plant safety could be increased by requiring the licensees to perform an analysis to identify and assess the potential hazards that exist at a facility. Furthermore, since risks and their prevention and mitigation may be of a competitive nature (e.g., fire protection versus criticality safety), it was recognized that an ISA would be useful to balance risks and their prevention/mitigation to achieve the best overall plant safety level.

In summary, the results of an ISA consist of an identification of potential events that are unacceptable, the consequences of the events, and the controls (i.e., the site, systems, structures, equipment, components, and human actions) relied upon to prevent the events from occurring or to reduce their consequences.
2.1 ISA Methods

There are a number of ISA techniques that may be used. The following serves as a partial list:

1. Safety Review
2. Checklist Analysis
3. Relative Ranking
4. Preliminary Hazard Analysis
5. What-If Analysis
6. What-If/Checklist Analysis
7. Hazard and Operability Analysis (HAZOP)
8. Failure Modes and Effects Analysis (FMEA)
9. Fault Tree Analysis
10. Event Tree Analysis
11. Cause-Consequence Analysis
12. Human Reliability Analysis

The choice of a particular method or combination of methods depends on a number of factors, including the reason for conducting the analysis, the results needed from the analysis, the information available, the complexity of the process being analyzed, the personnel and experience available to conduct the analysis, and the perceived risk of the process.

One of the most important factors in determining the choice of an ISA approach is the information that is needed from the analysis. To satisfy NRC requirements as defined in proposed new 10 CFR Part 70, the applicant should choose a method that identifies specific accident/event sequences, in addition to the safety controls that prevent or mitigate their progress/consequences.

3.0 The Role of ISA in a Facility Safety Program

One of the results of an ISA is an identification of the controls, both engineered and administrative, that are relied upon to limit or prevent event sequences or mitigate their effects. The identification of controls, however, is not sufficient to reasonably assure an adequate level of safety. In addition, the controls must be available and reliable to assure that, when called upon, they are in place and will operate properly. Therefore, the applicant is required to address the following program elements in the application:

1. Site-wide safety procedure development, review, approval, and implementation
2. Training
3. Maintenance, testing, calibration, and surveillance
4. Management of change (configuration control)
5. Quality assurance
6. Human factors
7. Audits and inspections
8. Emergency planning
10. Event response

The importance of these elements cannot be overstated. An ISA may be capable of identifying potential event sequences and the controls needed to prevent/mitigate them, but it cannot assure effective implementation of the controls and their proper operation. Without an effective safety program in place, the safety of a facility cannot be reasonably assured.

4.0 Unacceptable Consequences

The unacceptable consequences of concern (within NRC’s regulatory authority) are those that would result in the exposure of workers or members of the public to undue risk from radiation and certain chemicals at NRC licensed fuel cycle facilities. The mechanism for such exposure could be a release of radioactive material or an inadvertent nuclear chain reaction involving special nuclear material (criticality). The release of hazardous chemicals is also of regulatory concern but only to the extent that such hazardous releases result from the processing of licensed nuclear material or have the potential for adversely affecting radiological safety. The U.S. Occupational Safety and Health Administration and the Environmental Protection Agency are responsible for regulating other aspects of chemical safety and industrial safety at NRC licensed facilities. The following facility conditions are currently being considered for the proposed new Part 70:

1. For normal and off-normal conditions, the requirements of 10 CFR Part 20 are satisfied and unplanned nuclear criticalities are avoided.

2. For accident conditions, no member of the public receives a total radiation dose of (to be determined) rem to the whole body, an intake of (to be determined) milligrams of uranium in a soluble form, or an intake within any 30 minute period of (to be determined) milligrams of hydrogen fluoride.

For these conditions, the applicant/licensee must provide reasonable assurance that events leading to these consequences are not likely to occur.

5.0 Level of Protection/Graded Approach

In developing a safety regulation, one of the most crucial issues is that of identifying an adequate level of protection that will "place no undue risk to the public." In the formulation of the proposed new Part 70 and this issue is being addressed by delineating the level of protection that the staff concludes to be necessary for a licensee to operate safely. Currently, both deterministic and probabilistic methods are under consideration.

To provide reasonable assurance that the event sequences/consequences
identified by the ISA are prevented/mitigated, the NRC staff has developed a methodology ("graded approach") by which the level of protection is commensurate with the level of risk that exists for a given activity at a facility. In the graded approach to risk, three levels of consequences are being considered for radiological releases:

1. Those below 10 CFR Part 20 limits
2. Those greater than 10 CFR Part 20 limits, but less than protective action guidelines
3. Those greater than protective action guidelines

This graded approach to safety also applies to chemical hazards. (Essential programs needed to ensure safety at nuclear fuel cycle facilities, e.g., maintenance requirements, training requirements, etc., would also be graded commensurate to the risk.)

For event sequences that result in criticality, the staff proposes the use of the "double contingency" criterion which has been a standard of fuel cycle facility regulation. Essentially, the criterion requires that at least two unlikely and independent failures are necessary to enable the occurrence of a criticality. This has been adopted by industry in ANS Standard B.1 and has been the NRC/NMSS regulatory practice for several years.

For event sequences other than criticalities, such as those resulting in radioactive releases or other hazards associated with radioactive releases, and where the expected unmitigated consequences would be equal to or greater than the previously noted "protective action guidelines", the standard of reasonable assurance of "double protection" is under consideration as the criteria for the highest risk category. Double protection is defined as a requirement that there be at least two unlikely failures of barriers/controls prior to an unmitigated radioactive and/or toxic release. Unlike the case of double contingency, double protection affords the applicant the opportunity to utilize one control/barrier to mitigate the consequences of the accident sequence rather than to prevent the occurrence. Finally, in both the double contingency and double protection approaches, the quality level of the controls for high risk would be high. (See Section 6.0)

For those event sequences and associated consequences (other than criticalities) that result in less than the protective action guideline, but which are expected to exceed the requirements of 10 CFR Part 20, single protection (minimum) prior to an unmitigated release is under consideration. In this case, the quality level of the implemented barriers/controls would be accepted as a function of the level of the potential consequences and the actual number of controls in place in excess of the acceptable single control.
These cases would be determined during the staff's review of an application to provide the applicant reasonable flexibility in choosing and justifying a set of safe controls of acceptable quality, availability, and reliability.

6.0 Adequacy of Controls

An associated element in identifying a level of protection commensurate with the risk of the activity is the identification of criteria for the controls that are utilized to define a given level of safety. The staff is addressing this issue by developing a Standard Review Plan (SRP) which defines acceptance criteria for various controls utilized to provide adequate protection. For example, for nuclear criticality, the acceptance criteria for concentration is delineated in the SRP as:

The use of concentration as a control parameter for criticality control is acceptable if:

1. The solubility limits of the SNM composition are demonstrated.
2. Conditions which may affect the solubility are evaluated and controlled in accordance with the acceptance criteria for processes utilized as criticality controls.
3. Possible precipitating agents are identified to the operators through procedures and appropriate precautions taken to assure that they are not introduced.
4. A positive means of preventing unwanted transfers is provided if a possibility exists for precipitating agents to be transferred via connected processes. (The mechanisms that are evaluated for possible inadvertent transfer are evaluated: mechanical, chemical, and or thermal energies.)
5. An appropriate safety margin is established utilizing either experimental data or data derived from validated analytical methods, in accordance with the acceptance criteria for analytical methods.
6. The safety margin is demonstrated to be sufficient such that adequate controls are in place to control the quantity of the precipitating agent or change in process variable (i.e., pH, temp), that would be necessary to over-concentrate the solution.
7. Full reflection is utilized in deriving the appropriate limits unless controls are implemented in accordance with the acceptance criteria for neutron reflection.
8. Tanks containing the solution are normally closed. Supervisory personnel are required to be present when tanks are opened.
9. Sampling programs to measure concentration utilize a dual analysis and require supervisor approval before transferring solution.

10. Instrumentation utilized to measure the concentration is in accordance with the acceptance criteria for instrumentation.

Applicable acceptance criteria are also being considered for controls in other safety disciplines.

Another important aspect in delineating the adequacy of a control utilized to provide a given level of protection is the quality level of that control. The staff is addressing this issue by formulating an approach whereby the quality of specified barriers/controls will be acceptable at two different levels. NRC's approval of a particular level will be dependent on the safety importance of the control. The high quality level, as discussed in the above paragraphs, is in accordance with applicable national standards, such as ANS, ANSI, ASME, ASTM, IEEE, etc., and controlled within the applicant's own facility-wide overall quality assurance program. The other quality level may be an industrial or commercial grade level, but, as noted above, specific proposals would be considered during the review of applications.

7.0 Conclusion

The NRC is utilizing a risk-based approach in rewriting its Nuclear Materials Regulation, 10 CFR Part 70. This approach requires the licensee to perform a comprehensive safety analysis of the facility to identify potential hazards. In addition, applicants are required to implement appropriate levels of protection that are commensurate with the potential risk.
Licensing of the Sellafield MOX Plant SMP

By

Dr R J Page
Mr A J Tilstone

Paper Presented to

OECD Nuclear Energy Agency
Committee on the Safety of Nuclear Installations
Working Group on Fuel Cycle Safety
Topical Meeting on the Safety of Nuclear Fuel Cycle

Cadarache, France

20 - 21 September 1994
1. **INTRODUCTION**

It is BNFL Company Policy that inter alia:

'The Company will as a minimum conduct its activities within the framework of all relevant legislation'.

It is therefore important to have a general understanding of the UK Regulatory System to understand how a commercial scale MOX facility would be developed and licensed by BNFL in the UK.

This note is in two parts; it first describes the major legislation relevant to the licensing and operation of a MOX facility at Sellafield and provides an insight into how the UK Regulators operate. In the second part it describes how the BNFL Project arrangements and safety management systems have been developed to address the Companies legal obligations with specific reference to the Sellafield MOX Plant (SMP) project.
2. UK LEGISLATION AND REGULATION

2.1 Introduction

British Nuclear Fuels legal obligations result from two different types of legislation.

a. That which applies to any industrial installation in the UK. This covers the requirement for:

- the Health and Safety at Work etc. Act
- the Control of Substances Hazardous to Health (COSHH) Regulations
- planning permission
- approval under the Building Regulations
- the Factories Act and associated Regulations
- the Control of Pollution Act

b. That which applies to nuclear installations or to people dealing with radioactive materials.

In addition to the above, the UK nuclear industry is subject to:

c. European Union Requirements

d. International Atomic Energy Agency Regulations
It is not proposed to expand further on the 'conventional' legislation referred to in (a) except in relation to the Health and Safety at Work etc. Act which provides the underpinning legislation for the majority of safety regulation in the UK including the licensing of Nuclear Installations, but to concentrate on those aspects of legislation, regulation etc., which are specific to the nuclear industry.

2.2 Conventional Safety Legislation

Health and Safety at Work, etc Act 1974

A committee was appointed by the UK Government in 1970 to consider the whole range of law relating to occupational health and safety. It reported in 1972 and fundamentally criticised the existing legal framework. In particular it felt that:

1. the mass of existing legislation should be replaced by one all-embracing statute;

2. the law ought to lay down easily understood broad principles, and not a complex mass of detailed rules;

3. prosecution was not the best primary sanction;

4. the law ought to protect everybody, and not only employees;

5. there was too much emphasis on technical safety standards, and not enough on management's responsibility to provide systems of work which were actually safe when operated; and

6. the workforce ought to be involved in the safety effort.

The Government introduced legislative proposals based on this report in May 1973 and these were passed into UK law on 31 July 1974.

The Act established the Health and Safety Commission and the Health and Safety Executive. It makes provision for securing the health, safety and welfare of persons at work, for protecting others against risk to health or safety in connection with the activities of persons at work, and other activities related to health and safety. Under this Act the mass of detailed and technical legislation, which was administered and enforced by a variety of statutory agencies, was replaced by a simpler, coherent and co-ordinated body of regulations supported by practical guidance, and coming under the scrutiny of one central policy-making and enforcing body. That body is the Health and Safety Commission (HSC) with, as its enforcing arm, the Health and Safety Executive (HSE).
2.3 Nuclear Safety Legislation and Regulation

The Development of Nuclear Legislation in the UK

Regulation of nuclear energy is prescribed through Act of Parliament, but there is no single Act in the UK which covers all aspects of control. Subsequent to the start of the civil nuclear power programme and to some extent in response to the Windscale fire of 1957, legislation was enacted to give powers for the regulation of nuclear safety (Figure 1).

The Secretary of State for Trade and Industry has the general responsibility to parliament for the formulation of energy policy in England and Wales, whereas the responsibilities of the Secretaries of State for Employment and for the Environment include those for general and nuclear safety from operating facilities and safety of the general public from discharge and disposal of radioactive waste respectively. Because of the differing Governmental Department responsibilities in the UK, Regulation of nuclear fuel cycle activities is undertaken in two main areas. The first, nuclear safety including control over design, construction, commissioning, operation and maintenance of plant and facilities is regulated by the Health and Safety at Work etc. Act 1974 and its relevant statutory provisions. The second area, control over accumulation, discharge and disposal of radioactive waste, is regulated by the Radioactive Substances Act 1960. In the first area control is exercised by the Nuclear Installation Inspectorate (part of HSE) while in the second area control is exercised by the Authorising Departments and principally by Her Majesty’s Inspectorate of Pollution.

Regulation and Control of Nuclear Safety

The main legislation governing the safety of nuclear installations is the Health and Safety at Work etc Act 1974 (HSW Act) and the relevant statutory provisions of the Nuclear Installations Act 1965. Under these Acts no site may be used for commercial nuclear installations unless a Nuclear Site Licence has been granted to a corporate body by the HSE (Figure 2). HSE is the body empowered to take independent action on enforcement of safety legislation in the UK although it takes general policy instructions from the Health and Safety Commission (HSC). Her Majesty’s Nuclear Installations Inspectorate (NII) is that part of HSE responsible for administering and enforcing the licensing function set out in S1-S5 of the Nuclear Installations Act 1965 and for ensuring that any installation is operated safely both for the workforce and the general public.

The Site Licence is the legal document used by the NII to regulate nuclear safety. In addition, NII uses the specific requirements of Ionising Radiations Regulations 1985 to control exposures to ionising radiations (The IRRs 1985
enact in UK Legislation the European Council Directives of 90/836 and 84/467 Euratom).

The Site Licence (Ref 1) is a relatively small document and has the same general content for all UK installations; both power reactors and fuel cycle facilities. Typically for fuel cycle facilities, the Site Licence covers the whole site and is not specific to individual facilities.

Conditions attached to Site Licences specify requirements of HSE (NII) for the Licensee (BNFL) to make and implement 'adequate arrangements' relating to design, construction and installation of plants or processes. Similar conditions relate to commissioning, plant operations, maintenance, decommissioning and quality assurance arrangements. The Licence allows NII to intervene if arrangements are inadequate or require their approval of arrangement if considered appropriate. It is under these conditions and the Licensee's arrangements that NII exercise a number of powers controlling the activities of licensees through legal devices such as Consents, Approvals, Directions, Specifications or Agreements.

In regulating the activities of the licensees, the NII employs two main techniques. Firstly, by direct inspection of plant and working methods on site and if necessary manufacturers' works. Inspections are carried out in the main by designated inspectors to a planned programme. This planned programme is supplemented by inspection using multi-disciplined teams of inspectors to focus in depth on significant aspects of a plant or its associated activities. The second activity is assessment of licensee proposals (eg the proposal to build a new MOX plant). Essentially, before a licensee carries out an activity which has nuclear safety implications for the first time or wishes to modify existing arrangements, he must first submit proposals to the NII. The NII will assess these licensee proposals using as a guide its Safety Assessment Principles (SAPs). Assessment involves the examination of safety case documentation by a multi-disciplined team of NII Inspectors. Each Inspector, working within his own discipline, examines the relevant parts of the safety case. These SAPs, which have been recently updated, are published and provide guidance to assessors. They are not prescriptive in the sense that they represent a plant or product specification, rather they set the safety standards for civil nuclear installations. They reflect HSC policies, regulatory requirements and best engineering practice aimed at achieving a tolerable level of risk that is As Low As Reasonably Practicable (ALARP) (a UK legal term that is in practice equivalent to ALARA) based on the principle of defence in depth.

The current version of the SAPs (Ref 2) are much more wide ranging than previous versions and take account of the conclusions of recent nuclear Public Inquiries. The SAPs contain a total of 333 individual principles which address:
- Basic safety standards and criteria to be applied to individual plants.
- Safety assessment methodologies (deterministic and probabilistic).
- Design and Engineering covering all relevant engineering disciplines.
- Quality Assurance and general management arrangements.
- Siting of new facilities.

Finally, outside the day-to-day regulatory activities, there is the Advisory Committee on the Safety of Nuclear Installations (ACSNI) whose main objective is to provide the Health and Safety Commission and Ministers with independent and technically competent advice on major policy issues affecting safety of nuclear plants.

Accumulation and Disposal of Radioactive Wastes and Effluents

The Radioactive Substances Act 1960 regulates the keeping and use of radioactive material and makes provisions as to the accumulation and disposal of radioactive waste. The Act prohibits the accumulation and disposal of waste except in accordance with Certificates of Authorisations granted by the Secretary of State for the Environment and the Minister of Agriculture Fisheries and Food (Figure 3). Separate Authorisations are held by BNFL, for example, for the discharges of liquid and aerial effluents from the Sellafield Site and the disposal of solid low level wastes at Drigg. These Authorisations set down specified limits for waste disposal from (or in the case of a solid waste disposal facility to) the Site.

The Radiochemical Inspectorate (RCI) Branch of Her Majesty's Inspectorate of Pollution (HMIP) is that part of the Department of the Environment responsible for administering this Authorising function in collaboration with the Ministry of Agriculture, Fisheries and Food (MAFF).

The UK Government document Cmnd 6820 widened the responsibilities of the Department of the Environment for the overall development of radioactive waste management policy over and above the registration and authorisation requirement of the 1960 Act.

Although Authorisations are not required prior to active commissioning of any new facility there is consultation between HMIP and the NII in order to ensure that full account is taken both of the implications of waste management for site safety and of the waste management implications of proposals submitted to NII by operators of licensed nuclear sites. NII has undertaken that it will not give agreement under site licences for any proposals which are contrary to the national waste management strategy.
3. LICENSING OF A NEW SELLAFIELD MOX PLANT

3.1 Formal Safety Arrangements and Licensing Submissions

From the previous description of UK legislation it may be seen that the regulatory requirements for a new facility arise mainly from the Nuclear Site Licence and the conditions attached to it. It should, however, also be noted that NII will not agree to any new facility being constructed unless the DOE are happy with any associated waste management activities. Authorisations are also required prior to active commissioning to allow for the discharge of any radioactive effluents generated by the process.

The main feature of the UK System is that it is not in general prescriptive but it identifies those areas that may affect the safety of the workforce and the general public and requires the licensee (eg BNFL) to develop arrangements in these areas to adequately address safety. That is to say the system defines what has to be addressed but does not prescribe how. It allows the licencees to develop arrangements that are appropriate to the nature of their business and the nature of the risk this entails. The arrangements described here are therefore specific to BNFL and its fuel cycle business, but are generally compatible with those employed by other nuclear licensees in the UK.

The development of an overall Company Safety Management System within BNFL is the responsibility of BNFL’s Health and Safety Directorate (HSD). HSD is independent of BNFL’s operating Divisions. The Company Safety Management System takes due account of the NII’s SAPs and includes definition of the overall organisation and responsibilities for safety within BNFL and the numerical safety standards, criteria and principles against which new facilities will be developed. These standards and criteria address both normal operating and potential fault conditions and define:-

- Engineering best practices (these then form the basis of the detailed Company Engineering Standards).

- Normal operational dose targets to the workforce and the general public (arising from effluent discharges).

- A range of dose vs frequency bands that address potential fault conditions (probabilistic criteria).

- Deterministic and probabilistic criteria for external hazards (eg seismic, aircraft crash etc).
The underlying principles are derived from the NII SAPs and take due account of HSE's advice in the document Tolerability of risk from Nuclear Power Plants (Ref 3) ICRP's recommendations in Publication 60 (Ref 4). They equate to tolerable levels for the risk of death from accident for the Sellafield site as a whole of $10^{-6}$ y$^{-1}$ and $10^{-5}$ y$^{-1}$ for the most affected members of the general public and the workforce, respectively. Obviously as off site risks may be additive the tolerable risk from individual facilities is a small factor of the site risk.

The Site Licence Regulations are developed by individual operating Divisions and are specific to each Division. They specifically define the arrangements being implemented by the Division to address each individual condition attached to the Site Licence and have been approved by the NII. The Company Safety Management System and the Site Licensing Regulations (SLRs) together provide the basis of BNFLs working arrangements for the Licensing of any new facility. This is shown schematically in Figure 4. Working procedures (for example the SMP project procedures) are developed within the framework provided by these documents. With respect to any new facility that has nuclear safety implications, including the new Sellafield MOX Plant, the SLRs required that BNFL must obtain NII's 'Agreement' prior to commencing:

- construction
- inactive commissioning
- active commissioning
- operation.

In order to support BNFL's requests for such 'Agreements', the SLRs also require BNFL to perform detailed safety assessments and prepare formal safety reports at a number of key stages in the development of a new facility. For SMP, the SMP project manager is responsible for producing the formal safety reports and is helped in this task by a specialist safety assessment team who are part of the SMP task force.

The principal safety reports leading to the licensing and operation of a typical new Sellafield facility are:

1. Preliminary Safety Report (PSR) in support of the conceptual design. The PSR is optional and is not a formal requirement of the licensing process. It does however allow BNFL to obtain the NII's views of the potential licensability of the proposed new facility before significant design costs are incurred. If a PSR is submitted NII will either 'acknowledge' the document as providing a suitable basis for further design or 'object', in which case major re-design would be required.
Internally to BNFL, the main function of the PSR is to classify the safety significance of the proposed facility, to provide a safety specification against which the design can develop and to give confidence that the project cost estimate and programme contains adequate provisions for the development, installation and licensing of safety systems.

In the case of the SMP, a PSR was not prepared due to programme constraints which required the early preparation of a more detailed PCSR (Pre Commencement Safety Report, see below). The PCSR fully addressed all of the issues that would otherwise have been covered by the PSR. A simple safety statement was however issued to NII in June 1993 which provided NII with basic information on the SMP project and the proposed safety management arrangements. This ensured that the NII assessors were up to speed on the SMP project prior to them receiving the first PCSR and ensured that their consideration of the PCSR was as efficient as possible hence minimising the potential for delays.

2. Pre Commencement Safety Report (PCSR) in support of the commencement of construction. The PCSR provides a full safety justification of the proposed facility prior to the commencement of any construction work. Typically, the PCSR should be submitted to NII 6 months prior to start of construction to allow NII to assess the case and to complete any necessary negotiations prior to their agreement to commence construction. In many cases where it is necessary to construct a new facility to a tight programme, it is not possible to provide full safety justification addressing all construction and installations prior to commencing the initial civil construction work. Generally the design is not sufficiently detailed in all areas at such an early stage. In such cases the PCSR may be phased with separate versions supporting successive construction phases.

For the Sellafield MOX Plant it was proposed (and accepted by NII) to provide two versions of the PCSR. The first PCSR was issued to NII in August 93 supported the commencement of the civil construction and demonstrated that the civil design incorporates all the necessary safety provisions (eg shielding, seismic performance etc) without unduly constraining the subsequent safety performance of mechanical, electrical and control systems. Agreement to start construction was obtained from NII in March 1994. The second PCSR was issued to NII in June 1994, provides the full design safety justification and supports the commencement of mechanical, electrical and instrumentation (ME&I) installation, NII agreement to commence ME&I installation is expected in September 1994.
3. Pre Commissioning Safety Report (PCmSR) in support of commencement of inactive commissioning. The PCmSR updates the PCSR to the as-built status. It is the definitive design safety justification. The PCmSR also provides the basis for defining the safety systems that support the safety justification and require to be commissioned during inactive safety commissioning.

The full range of commissioning tests required to support the safety case contained within the PCmSR are defined in the Safety Commissioning Schedule (SCS). This is submitted to the NII together with the PCmSR and the commissioning manual describing quality arrangements in the order of 6 months prior to the start of commissioning to allow NII to assess the adequacy of BNFL's proposal and complete any necessary negotiations prior to their agreement to commence commissioning. The SMP PCmSR is scheduled for issue to NII in May 1996.

4. Plant Safety Case (PSC) in support of active commissioning and the initial operation of the plant. The PSC updates the PCmSR to take account of any modifications arising during inactive commissioning. The PSC also defines the formal operating arrangements including definition of 'Safety Mechanisms', Operating Rules and the Maintenance Schedules required by the SLRs to support the safety case for an operating facility. The SMP PSC is scheduled for issue to NII in March 1997.

The PSC is typically updated after the first years operation and every five years thereafter to reflect operating experience and changes to plant, equipment and operating procedures.

The overall licensing requirements and their relationship to the main project phases are shown in Figure 5 and the SMP key dates are summarised in Table 1.

3.2 The Safety Assessment Process being applied to SMP

Within BNFL the safe design of facilities is achieved via two complimentary routes:

1. Deterministic safety by the design of the plant against a comprehensive range of international, national and Company Standards as defined by the Company Safety Management System. Central to this system are a number of Company engineering standards which take due account of NII's requirements as defined in their SAPs and define the requirements for safety systems on the basis of the level of potential radiological consequences. Where appropriate, the detailed design proposal is
subject to independent peer (engineering) review to confirm the relevant standards have been applied and approval is required from the peer review prior to commencing work on site.

2. Probabilistic Safety Assessment (PSA) applied to design proposals and operational arrangements to demonstrate that the assessed risk is acceptable by comparison with Company criteria.

PSA is the main technique applied to the review, assessment and justification of any new facility in BNFL and provides the basis of the numerical assessments contained within each of the above safety reports. A new facility is subjected as far as is practicable to a full probabilistic assessment at each stage (ie each safety report) of the projects life cycle. The scope of the study applied at each stage is shown in Figure 6.

At the PSR (and where prepared the Civil PCSR) stage, the PSA study concentrates on more significant hazards that may occur identified by HAZOP I study and are based on bounding estimates of the potential on and off-site radiological consequences and initiating event frequencies. The assessments provide a means of developing a safety schedule which as far as practical, defines the requirements for the principal safety systems in terms of function and where appropriate, integrity. This then provides a direct link to the relevant Company Engineering Standards and the deterministic safety case.

At the PCSR stage the design is more developed but the final working drawings are not generally available. The PCSR therefore provides a detailed probabilistic safety justification based on statements of design intent and assumptions of operational procedures, etc. All hazards with significant radiological consequence (ie those capable of leading to an off site consequence >5μSv or a building emergency) identified by a detailed HAZOP 2 study are addressed. As far as practicable, full frequency and consequence assessments are performed and the design shown to comply with the numerical Company Safety Criteria and Standards. There may however still be a number of outstanding issues identified which require resolution prior to operation.

The PCSR also contains a fully worked up version of the safety schedule which summarises the probabilistic and deterministic safety justification in a tabular form and identifies all the significant engineering provisions and operating assumptions necessary to achieve the necessary level of safety. The safety schedule presents the detailed safety information in a form that can be readily understood by the plant engineers and future operators (and the Regulators) and is an important aid in ensuring the design and operating requirements are properly understood and implemented. Also by providing the basis of the safety case in a concise manner the safety schedule is an important component of change control.
At the PCmSR stage the design is complete and the final working drawings available. The PSA contained within the PCSR is updated to reflect the as-built standard and any safety significant changes justified. Where appropriate, the hazard assessments are more detailed and take account of any additional hazard identification exercises, eg task analysis of provisional operating procedures. At this stage provisional 'Safety Mechanisms', 'Operating Rules' and Maintenance Requirements are identified.

The PCmSR identifies all safety related equipment and its function (ie the safety schedule) and provides the basis for defining the safety tests that must be performed during commissioning to demonstrate the safety performance of the plant prior to commencing active operation. These safety tests including cross reference to PCmSR, are contained within the Safety Commissioning Schedule (SCS).

The PSC updates the PCmSR to take account of commissioning and defines the formal operating arrangements including definition of 'Safety Mechanisms, 'Operating Rules' and maintenance schedules as required by the SLRs. The PSC is updated periodically (typically after 1 years operation and each subsequent 5 years) to reflect operating experience and changes to plant, equipment or operating procedures.

3.3 Influence of the Safety Assessment Process on SMP

Implementation of the safety assessment process as defined in the previous section of this paper has enabled safety considerations to be an integral component of the design process from a very early stage of project. This has enabled a number of potential hazards to be removed or minimised and has reduced the requirements for additional safety protection systems which would have added to both complexity of design and operating cost of the plant. This is best illustrated by considering how the potential hazards of routine operational dose uptake, criticality and loss of containment have influenced the design of SMP.

Operational Dose Uptake

An iterative strategy has been applied in order to develop a plant design which would result in an acceptable level of dose uptake. This strategy is illustrated schematically in Figure 7. Implementation of this strategy revealed at a very early stage of the project design that the use of existing MOX plant technology, which involved a high degree of manual operations particularly in fuel assembly operations, was unacceptable in terms of dose uptake for a production plant with the throughput capability and flexibility of SMP. Therefore the design of SMP has been developed largely around the concept of remote
operation with appropriate local or bulk shielding provided around the process stages with relatively high dose rates. This is best illustrated in the fuel assembly areas of SMP where the whole fuel assembly line operations are remotely operated and are enclosed in a concrete shielded room of appropriate thickness.

One of the major components of the operational dose uptake assessment process has been the development of a detailed plant operational model broken down to an individual task level. Preparation of this model was commenced very early in the project design and drew on the extensive experience gained by BNFL over many years of designing, operating and decommissioning plutonium plants, and more recent experience of operating MOX fuel fabrication plants. This experience provided valuable data on both the manpower requirements and task durations for both process and maintenance operations, and on the main potential short and long term sources of operational exposure and how these exposures could be adequately controlled.

The SMP dose uptake assessments are therefore based on very robust base data and a detailed estimate of the dose uptake for SMP indicates that the annual average dose uptake for the SMP workforce including maintenance operations will be less than 5mSv y\(^{-1}\) at full capacity and using high burnup long cooled Pu.

**Criticality**

Criticality was recognised as one of the main potential hazards associated with SMP from the beginning of the project design. As a result considerable effort has been concentrated on the development of a plant design which has an acceptable criticality safety case, but which is not constrained in terms of its flexibility to process a wide variety of feed material and manufacture a wide range of fuel assemblies. In order to satisfy this requirement the criticality assessments of the SMP design have been based on the most criticality reactive feed materials and fuel assemblies which effectively bound all other potential feeds and products. Given this design basis the principal objective of the design was to make plant inherently safe where practicable for normal operations and potential fault conditions. Only where this objective is not practicable are additional safety protection systems provided. This obviously enhances the robustness of the safety of the plant and reduces the requirement for potentially complex safety protection systems which add to both plant design and operational (e.g. maintenance) costs. An example of where this objective has influenced equipment design include the MOX powder blender in the main powder processing area which has been designed as a thin ‘slab’ type blender (i.e. safe shape) compared with conical blenders utilised on other similar plants.
One of the major constraints in developing a criticality safety case for SMP has been the limited reliability data which is accepted by UK Regulators for software based control systems. As described above, SMP is a remotely operated plant relying extensively on such a system for control of the process. It has been identified in the extensive hazard identification exercises (eg HAZOP studies) carried out on the SMP design, that failure of multiple functions within the control system could give rise to potential criticality incidents in certain process vessels which are not practicable to remove by equipment design (ie the vessels would become too small to maintain required throughput). Therefore although the design of SMP has minimised the requirement, a small number of safety protection systems have been incorporated into the SMP design to prevent potential criticality incidents initiated by multiple failure of functions in the control system. These safety protection systems are diverse and are engineered to be fully independent of the control system to avoid potential common mode failures. The potential for criticality incidents in the SMP plant has been calculated as less than $10^{-7}$ y$^{-1}$. This represents a considerable design achievement given the limited reliability data which is accepted by the UK Regulators for software based control systems.

**Loss of Containment**

Most of the SMP processing operations up to the loading of filled fuel rods into fuel magazines and assemblies are carried out in glovebox containment. A number of design features have been incorporated into the design to prevent potential loss of containment. These include:

- prevention of potential impact damage to gloveboxes due to mechanical failure of in-box components. This includes the use of protective sheaths around drive shafts, physically restricting the movements of equipment, positioning glovebox windows in appropriate locations, away from potential impact areas;

- prevent glovebox pressurisation by fitting process gas supplies with flow restrictors and additional isolation valves which automatically trip if glovebox depression is lost (eg upon a failure of the glovebox extract system);

- utilising vortex amplifiers to provide emergency breach protection should glovebox containment be breached (eg glove accidentally removed).

Active material handled outside gloveboxes is contained in containment barriers of appropriate integrity (eg PuO$_2$ cans, fuel cans). The handling systems for these materials takes due account of the integrity of the
containment barriers, by limiting the forces applied to the containment and minimising potential drop heights.

4. CONCLUSIONS

A MOX fabrication facility at Sellafield is being licensed like any other commercial nuclear plant in the UK under the conditions of a Nuclear Site Licence. There are no special requirements for a MOX plant. The construction, commissioning and operation of the facility will be in line with 'arrangements' developed and implemented by the 'Licencee' as required by the Nuclear Site Licence issued by NII. Discharge Authorisations are also required to allow discharge of aerial and liquid effluents prior to active commissioning. All the necessary arrangements have been developed and are in place and accepted by NII under the existing Sellafield Nuclear Site Licence to support the construction and operation of the existing small scale MOX Demonstration Facility (MDF) and the full scale Sellafield MOX Plant (SMP).
<table>
<thead>
<tr>
<th>Date</th>
<th>Event Description</th>
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<tbody>
<tr>
<td><strong>JUNE 1983</strong></td>
<td>SAFETY STATEMENT ISSUED TO THE REGULATORS</td>
</tr>
<tr>
<td><strong>AUGUST 1993</strong></td>
<td>PCSR (CIVIL) ISSUED TO REGULATORS</td>
</tr>
<tr>
<td><strong>MARCH 1994</strong></td>
<td>REGULATORY AGREEMENT TO COMMENCE CIVIL CONSTRUCTION</td>
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<tr>
<td><strong>JUNE 1994</strong></td>
<td>PCSR (M,E&amp;I) TO BE ISSUED TO REGULATORS</td>
</tr>
<tr>
<td><strong>SEPTEMBER 1994</strong></td>
<td>REGULATORY AGREEMENT REQUIRED TO COMMENCE 1st PHASE M,E&amp;I INSTALLATION</td>
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<tr>
<td><strong>MAY 1996</strong></td>
<td>PCmSR TO BE ISSUED TO REGULATORS</td>
</tr>
<tr>
<td><strong>MARCH 1997</strong></td>
<td>PSC (ACTIVE COMMISSIONING TO BE ISSUED TO REGULATORS</td>
</tr>
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</table>
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   HSE, HM.

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4. ICRP Publication 60.
General and Nuclear Safety Regulation

(Principal Regulator NII)

Health & Safety at Work Act 1974

Nuclear Installation Act 1965

Site Licence

IRRs 1985

Radiological Protection Requirements

HMIP (Waste Mgt Issues)

Consultation on Waste Management Issues

Specific Licence Conditions

Key

ENNRI  Ionising Radiations Regulations

Figure 2
Radioactive Wastes and Discharges Regulation

**Disharges**

(Principal Authorising Departments HMIP and MAFF)

- Radioactive Substances Act
  - Discharge Authorisation
  - Specific Limits on Discharges

**Waste Management Strategy**

(Responsible Authorising Department HMIP)

- Command 6820
  - National Waste Management Strategy
    - Guidance Notes
      - Review of BNFL's Waste Management Proposals
        - Consultation on Safety Implications of Waste Management
          - NII
            - Nuclear Safety Regulations
              (Issue of Consents etc)

**Key**
- HMIP: Her Majesty's Inspectorate of Pollution
- MAFF: Ministry of Agriculture, Fisheries and Food
- NII: Her Majesty's Nuclear Installations Inspectorate
The Overall Development of Working Arrangements for a New Facility

- HSE Safety Policy
- Nuclear Site Licence
- BNFL Operating Division SLR's
- BNFL Project Specific Procedures etc
- NII Review/Approval
- BNFL HSD Company Safety Management System
- NII
- Safety Assessment Principles
Typical Flow Diagram
for Safety Review Assessment and Justification

- Design Safety Principles
- Previous Design Stage
- Current Design Stage Proposal
- Design Review Hazop etc Engineering Validation
- Recommendation Advice Modification
- NII Requirements
- Data Bases Methods etc
- Hazard Assessment
- Safety Standards and Criteria
- Comparison with Criteria
- Current Design Safety Viable
- Safety Spec for Future Eng Phases
- Outstanding Issues
- Operational Requirements
- Safety Justification
- Project Checking and Approval
- BNFL Peer Review
- MSC Consideration
- Issue to NII
- Discussion
- Agreement

Key
MSC Management Safety Committee
HAZOP Hazard and Operability Study
Figure 7

Iterative Strategy

Plant Design
Occupancy
Source Positions
Number of Staff
Source Assumptions
Shielding

Dose Rate and Dose Calculations

Dose Meets Targets
YES

Confirm ALARP
YES

Stop
Safety Regulations for Nuclear Fuel Facilities in Japan

Kenkichi HIROSE

Director
Nuclear Materials Regulation Division
Nuclear Safety Bureau
Science and Technology Agency
1. Outline of Safety Regulations

The nuclear fuel facilities that fall under safety regulations include milling and refining, manufacturing and reprocessing facilities, and other facilities that use nuclear-fuel materials and nuclear-source materials. Nuclear fuel cycles are composed of these fuel facilities together with nuclear reactors. The nuclear fuel facilities should be checked for the appropriateness of the design, construction process, and methods of operation and maintenance, during the preconstruction, construction stages, and the operation stage, respectively. The confirmation process is implemented according to "the Law for the Regulation of Nuclear Source Materials, Nuclear Fuel Materials and Nuclear Reactors." (note) (Ref. 1 ~ 5)

(1) Milling and refining facilities

Anyone who intends to establish a refining business other than the PHC (Power Reactor and Nuclear Fuel Development Corporation) should obtain the designation by the Prime Minister and the Minister of International Trade and Industry.

An administrative examination of the planned these business is to be conducted prior to the designation in terms of planned execution, technical capability, financial foundation, and disaster prevention. Prior to the designation of these business, the Prime Minister and the Minister of International Trade and Industry should ask for and respect the opinions of the Atomic Energy Commission and the Nuclear Safety Commission, regarding application of the requirements.

Prior to operation, the operator must get the approval of the Prime Minister and the Minister of International Trade and Industry for operator's safety regulations. Records documentation, waste disposal management, and reports of accidents and failures are also required from the start of operation.

(2) Manufacturing facilities (conversion, enrichment and fabrication)

Anyone who intends to establish a manufacturing business for nuclear fuel materials should get the permission of the Prime Minister.

An administrative examination of the planned manufacturing business is to be conducted in terms of manufacturing capability unduly excessive, technical capability, financial foundation and disaster prevention, prior to the permission.

Note) Safety regulations for mines of nuclear source materials are subject to "the Mine-Safety Law".
Prior to the permission for the manufacturing business, the Prime
Minister should ask for and respect the opinions of the Atomic Energy
Commission and the Nuclear Safety Commission, regarding application
of the permission requirements.

Subsequently after the acquisition of permission, the manufacturing
business is required to get the approval by the Prime Minister of
design and construction method prior to the start of construction.
The business is required to receive a inspection of facilities prior
to the start of operation as well. In addition, prior to starting,
the business should get the approval of the Prime Minister, regard-
ing operator’s safety regulations. The approval of the Prime
Minister concerning welding method together with a welding inspec-
tion certificate are required. Records documentation, safety
management, and reports of accidents and failures are also required
from the start of business.

(Ref. 6)

(3) Reprocessing facilities

Anyone who intends to establish a reprocessing business other
than the FNC or the JAERI (Japan Atomic Energy Research Insti-
tute) should obtain the designation by the Prime Minister.
FNC and JAERI should get the approval of the Prime Minister for
establishing a reprocessing facility.
An administrative examination of the planned reprocessing business
is to be conducted in terms of peaceful utilization, planned
execution, technical capability, financial foundation and disaster
prevention. An examination disaster prevention is also to be carried
out at the time of the approval.

The Prime Minister should ask for and respect the opinions of the
Atomic Energy Commission and the Nuclear Safety Commission (in the
case of granting approval, the Nuclear Safety Commission alone is
consulted), regarding application of the requirements prior to the
designation or approval of the reprocessing business.

Subsequently, after the acquisition of a designation or approval,
the reprocessing business is required to get approval of the
design and construction method, a inspection before use, and
the approval of operator’s safety regulations. Also, the ap-
proval of the welding method and a welding inspection certif-
icate should be required. In addition to these duties similar-
ly implemented for a manufacture, the reprocessing facilities
should pass all periodical inspection (once a year) after the
business starts.
(4) Facilities that use nuclear fuel materials and nuclear source materials

Any facility with an intention to use nuclear fuel materials, other than a miller, manufacturer and reprocessor, or operator of nuclear reactors, should get the permission of the Prime Minister. However, according to the Government Ordinance under the Law for the Regulations of Nuclear Source Materials, Nuclear Fuel Materials and Nuclear Reactors (hereinafter referred to as "Ordinance"), natural uranium, depleted uranium and their compounds or materials containing them, which can be used as nuclear fuel in nuclear reactors and that have a uranium content of 300 g or less, are not subject to the requirements of permission. Thorium and its compounds or materials containing it, which can be used as nuclear fuel in nuclear reactors and which have a thorium content of 900 g or less, are not subject to the requirements of permission for use, either.

An administrative examination of the planned use of nuclear fuel materials is to be conducted in terms of peaceful utilization, planned execution, disaster prevention and technical capability, prior to the permission.

Records documentation, observance of technical standards for use, and reports of accidents and failures are required for users from the start.

In addition, passing the facilities inspection of the construction method and welding, and approval of operator's safety regulations, and required for facilities possessing approval for use, where the following radioactive materials defined in Ordinance Article 16-2 are to be handled:

1. unsealed plutonium and its compounds, each with a plutonium content of 1 g or more.
2. spent fuel of 3.7 TBq (100 Ci) or more.
3. uranium hexafluoride with a uranium content of 1 t or more, and
4. liquid uranium and its compounds, each with an uranium content of 3 t or more. (Ref. 7)

Anyone who intends to use nuclear source materials should notify the Prime Minister, if the amount and radioactive level of the nuclear source material exceeds the specific value described in the regulations.

Records documentation, observance of technical standards for use, and reports of accidents and failures are also required for users of nuclear source materials from the start.
2. Formalization of Regulatory Guides for Licensing

Safety Regulation for nuclear fuel facilities as well as nuclear reactor facilities should be conducted based on the objective and reasonable evaluation standards. Therefore the Nuclear Safety Commission takes the responsibility of formulating various regulatory guides regarding nuclear fuel facilities. The Guides for nuclear fuel facilities has been discussed by the Special Committee on Safety Standards of Nuclear Fuel (established in November 1978) within the Nuclear Safety Commission. The Commission has established the "Basic Guide for Licensing of Nuclear Fuel Facilities" (adopted in February 1980, by the Nuclear Safety Commission) which provides the general principles for safety examinations of nuclear fuel facilities.

"Regulatory Guide for Licensing of Uranium Fabrication Facilities" (adopted in December 1980, by the Nuclear Safety Commission) is a guide for licensing review characteristics of uranium fabrication facilities. It is based on the principles for safety examinations of nuclear fuel facilities: "Reference Dose for Plutonium Intake in Relation to Site Evaluation of Nuclear Fuel Facilities, a dose mark of plutonium for determining the conditions at locations for potential nuclear fuel facilities in association with the incidence of emergencies (adopted in April 1983, by the Nuclear Safety Commission). The Regulatory Guide for Licensing of Reprocessing Facilities" (adopted in February 1987, by the Nuclear Safety Commission) is applied to reprocessing facilities where spent fuel generated by light water power reactors is reprocessed using the wet method (the PUREX process). The guide is provided, particularly for the targeting of civil reprocessing plants now under licensing review that may require the current safety examination. (Ref. 8).
3. Operation and Maintenance

For the operation and maintenance of nuclear fuel facilities, fundamental items, such as essential safety measures have been described in the Orders of the Prime Minister’s Office respective business types of facilities such as the manufacturing business of nuclear fuel materials, the reprocessing business of spent nuclear fuel, etc. in correspondence with individual facilities.

The Operator’s Safety Regulations approved by the Prime Minister (excluding the facilities using nuclear fuel materials and nuclear source materials below a specific amount) defines the organization for operation and maintenance and detailed rules for equipment operation, etc. Principal information, such as radiation control records should be submitted to the Minister of State for Science and Technology Periodically (excluding the facilities that use nuclear source materials). In case of incident, it is to be reported immediately, independent of the above routine report.

The operation and maintenance of nuclear fuel facilities has been depending on the level of required safety under the strict supervision of the Minister of State for Science and Technology. An expert officials on operation and maintenance is sent to reprocessing sites to check the actual status of the operation and maintenance. Thus, the organization for controlling operation and maintenance at nuclear fuel facilities has been significantly strengthened.

Under the Law for the Regulations, 2 reports related to troubles has been submitted to the authorities between April 1991 and March 1992. (Ref. 9)

4. Occupational Exposure Control

Radiation control in nuclear fuel facilities, as same as nuclear reactors, is comprised of radiation control of areas inside facilities and exposure control of personnel who enter the facilities.

The actual occupational exposure at all nuclear fuel facilities in 1991 were far below the dose equivalent limit. (Ref. 10)
5. Radioactive Waste Management

Radioactive waste management of nuclear fuel facilities applies to gas, liquid, and solid radioactive wastes.

The mean level or discharged amount of each radioactive material in radioactive waste in the form of gas and liquid is measured periodically or at each discharge, generally at the exhaust outlets. Diffusion effect during the outlet of the exhaust gas to the boundary of the nearby monitoring area may be accounted, in order to verify that the radioactive level is below the concentration limits and dose equivalent limits.

The radioactivity of solid waste and the activity per container of waste should be recorded at each occasion and the waste should presently kept in storage facilities under radiation control. In particular, high-level radioactive wastes generated in reprocessing facilities are to be stored in exclusive-use storage tanks after an evaporation and concentration process at this moment.

The discharged amount of radioactive wastes generated in the reprocessing facility in 1991 satisfactorily met the annual discharge limits. (Ref.11)

An environmental monitoring is required for reprocessing facilities to check periodically the radioactive levels in various environmental samples such as drinking water, vegetables, and marine water in their vicinity. This monitoring is implemented, in accordance with the Environmental Monitoring Program developed by the Central Committee on Evaluation of Monitoring of Environmental Radiation in the Nuclear Safety Commission. The results of the environmental monitoring are to be reviewed by the Central Committee. In December 1991, the Committee prepared a report "Results of Environmental Monitoring in the Vicinity of the Reprocessing Facilities of Power Reactor and Nuclear Fuel Development Corporations (1990)." The Central Committee did not notify the Nuclear Safety Commission of any problems.

(1) Current Status of the Nuclear Fuel Cycle Facilities in Rokkasho

① A commercial base for Japanese nuclear fuel cycle has been planned in Rokkasho, Aomori. (Ref. 12)

② Outline of the facilities

a. Uranium enrichment facilities
   Applicant: Japan Nuclear Fuel Limited
   Capacity: 1,500 ton SWU per year (at final stage)
   Permission of business: Aug., 1988
   Start of construction: Oct., 1988
   Start of the operation of 150 ton SWU per year: March 1992
   The operation of 300ton SWU per year: December 1992
   The operation of 450ton SWU per year: May 1993

b. Low-level radioactive waste disposal facility
   Applicant: Japan Nuclear Fuel Limited
   Final Capacity: $6 \times 10^3 \text{ m}^3$ (3 million drums with 2002 drum each)
   Application for $4 \times 10^4 \text{ m}^3$: April 1988
   Permission of business: November 1990
   Start of construction: November 1990
   Planned start of the operation: 1992

c. Reprocessing plant
   Applicant: Japan Nuclear Fuel Limited
   Reprocessing capacity: 800 ton U per year
   Spent-fuel storage capacity: 3,000 ton U.
   Application for licensing: March 1989
   Permission of business: December 1992
   Start of construction: April 1993

d. Waste management facility
   Applicant: Japan Nuclear Fuel Limited
   Final storage capacity: 3,000 pieces of verified high level waste
   Application for 1,440 pieces within above capacity: March 1989
   Permission of business: April 1992
   Start of construction: May 1992
   Planned start of operation: 1995

(2) Operation of the PNC Tokai Reprocessing Plant

① Reprocessed Cumulative Fuel is 680 ton U at March 1993.
   (Start of hot operation using spent fuel in 1977)

② Setting up testing facilities for recycle equipment - under inspection.
(3) Other Major Nuclear Fuel Facilities

1. PNC Ninmyo Toge Uranium Enrichment Demonstration Plant
   Start of full operation; May 1989 (200 ton SWU per year)

2. PNC Tokai MOX Fuel Production Plant
   Start of operation of FBR fuel line; April 1988. (Fuels for the
   FBR prototype reactor 'Monju')(Annual capacity: 5 ton MOX.)

3. JAERI Tokai NUCEF
   Construction of facilities for the research on criticality safety
2. Organization of Nuclear Safety Regulations

**Prime Minister's Office**
- **Nuclear Safety Commission (NSC)**
  - Committee on Examination of Reactor Safety
  - Committee on Examination of Nuclear Fuel Safety
- **Atomic Energy Commission (AEC)**
- **Science & Technology Agency (STA)**
  - Nuclear Safety Bureau (NSB)
  - Technical Advisors for Nuclear Safety
  - Nuclear Safety Policy Division
    - Office of Radioactive Waste Regulation
  - Reactor Regulation Division
  - Nuclear Materials Regulation Division
    - Office of Nuclear Materials Transport
- **Ministry of International Trade and Industry**
- **Ministry of Transport**
### 3. Regulatory System under the Law for Regulations

<table>
<thead>
<tr>
<th>Milling and refining</th>
<th>Manufacturing</th>
<th>Reprocessing</th>
<th>Use of nuclear fuel materials</th>
<th>Use of nuclear source materials</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Pre-construction stage</strong></td>
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<tr>
<td>Examination and review by Nuclear Safety Commission and Atomic Energy Commission</td>
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<td>Examination and review by Nuclear Safety Commission and Atomic Energy Commission</td>
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<tr>
<td>Approval of design and construction method</td>
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<tr>
<td>Approval of welding method</td>
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<td>Inspection of facilities</td>
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<td><strong>Operating stage</strong></td>
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<td>Notification of start of business (excluding PNC)</td>
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<td>Observation of technical standards</td>
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</tbody>
</table>
4. Safety Regulations for Reprocessing Facilities

(Preconstruction Stage)

Designation of Business (PHC and JAERI: Approval of Installation)

Application [Article 44] → Designation or Approval [Article 44] → Prime Minister

Report

Review by Science and Technology Agency
At the review, results of the site inspection, regulatory guides for licensing established by the Nuclear Safety Commission, etc. are referenced.

Requirements for Designation (Article 44-2)
1. Peaceful utilization
2. Planned execution
3. Financial foundation
4. Technical capability
5. Disaster prevention

Inquiry [Article 44-2] (Technical capability, Disaster prevention)

Nuclear Safety Commission

Direction of Review Report

Committee on Examination of Nuclear Fuel Safety

Design and Construction Method (Design and construction method are reviewed in terms of technical standards.)

Application [Article 45] → Approval [Article 45] → Minister of State for Science and Technology

Welding Method and Inspection (Approval of welding method; welding method is carried out based on technical standards)


Atomic Energy Commission
Articles in the Law for Regulations of Nuclear Source Materials, Nuclear Fuel Materials and Nuclear Reactors are put in the brackets.
## Outline of Nuclear Installations in Japan

(As of September 2, 1979)

<table>
<thead>
<tr>
<th>Object(s) of business</th>
<th>Minister responsible</th>
<th>Facilities under regulation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Milling and refining</td>
<td></td>
<td>1 facility (FNC, Kinya-kei)</td>
</tr>
</tbody>
</table>
| Manufacturing         | Prime Minister (Minister of State for Science and Technology) | • Fabrication to Fuel Assembly for LUR: 4 facilities  
• Uranium-fuel conversion for LUR: 2 facilities  
• Plutonium-fuel conversion facilities for a advanced thermal demonstration reactor (under construction, FNC Tokai-mura, Ibaraki) |

(Uranium enrichment)  
• Demonstration plant (FNC, Kinya-kei)  
• Commercial plant (Japan Nuclear Fuel Limited  
Rokkasho-mura, Kamikita-gun, Aomori) |

<table>
<thead>
<tr>
<th>Nuclear reactor</th>
<th>Research reactors</th>
<th>26 reactors (including 4 reactors under decommissioning)</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>Nuclear reactor</th>
<th>Reactors in R&amp;D stage</th>
<th>4 reactors (&quot;Hatsui&quot;, &quot;Togen&quot;, &quot;Honju&quot;, &quot;A advanced thermal reactor for demonstration (under planning)&quot;)</th>
</tr>
</thead>
</table>

| Commercial power reactors | Minister of International Trade and Industry | In commercial operation: 47 reactors  
Under construction: 6 reactors  
Preparing for construction: 2 reactors |

| Commercial ship reactors | Minister of Transport | No reactors corresponding to this classification |

| Reprocessing | Prime Minister (Minister of State for Science and Technology) | Tokai Reprocessing Facility (FNC, Tokai-mura)  
Commercial reprocessing facility (under licensing review, Japan Nuclear Fuel Limited, Rokkasho-mura, Kamikita-gun, Aomori) |

| Waste disposal | Waste burial | Low-level radioactive waste disposal facilities (Japan Nuclear Fuel Limited  
Rokkasho-mura, Kamikita-gun, Aomori) |
|----------------|-------------|-----------------------------------|

| Waste management | Radioactive waste management facilities (Japan Nuclear Fuel Limited  
Rokkasho-mura, Kamikita-gun, Aomori) |

| Disposal | Storage facility at the storage Engineering Research Center (under planning in FNC) |

<table>
<thead>
<tr>
<th>Use (Nuclear Fuel materials)</th>
<th>175 facilities</th>
</tr>
</thead>
<tbody>
<tr>
<td>Use (Nuclear source materials)</td>
<td>6 facilities</td>
</tr>
</tbody>
</table>
6. Nuclear Fuel Manufacturing Facilities

(As of June 30, 1993)

<table>
<thead>
<tr>
<th>Name of Establishments</th>
<th>Location</th>
<th>Licenced Date</th>
<th>Enrichment</th>
<th>Maximum Capacity</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>Japan Nuclear Fuel Co.</td>
<td>Yokosuka-city, Kanagawa</td>
<td>Aug. '68</td>
<td>Less than 5%</td>
<td>750 ton U/year</td>
<td>Fuel rod for BWR</td>
</tr>
<tr>
<td>Tokai Works, Mitsubishi Nuclear Fuel Co.</td>
<td>Tokai-mura, Naka-gun, Ibaraki</td>
<td>Jan. '72</td>
<td>Less than 5%</td>
<td>440 ton U/year</td>
<td>Fuel rod for PWR</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>475 ton U/year</td>
<td>Conversion for PWR</td>
</tr>
<tr>
<td>Tokai Works; Kumatoro Works, Nuclear Fuel Industry Co.</td>
<td>Tokai-mura, Naka-gun, Ibaraki</td>
<td>Sep. '78</td>
<td>Less than 5%</td>
<td>200 ton U/year</td>
<td>Fuel rod for BWR</td>
</tr>
<tr>
<td></td>
<td>Kumatoro machi, Senman-gun, Osaka</td>
<td>Aug. '75</td>
<td>Less than 5%</td>
<td>324 ton U/year</td>
<td>Fuel rod for PWR</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Sep. '72</td>
<td>Approximately 90%</td>
<td>475 kg U/year</td>
<td>Fuel plate for research reactors</td>
</tr>
<tr>
<td>Tokai Works, Japan Nuclear Fuel Conversion Co.</td>
<td>Tokai-mura, Naka-gun, Ibaraki</td>
<td>Sep. '80</td>
<td>Less than 5%</td>
<td>715 ton U/year</td>
<td>Conversion for BWR &amp; PWR</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Jun. '84</td>
<td>3%</td>
<td>3 ton U/year</td>
<td>Conversion for research reactors</td>
</tr>
<tr>
<td>Mingyotoge Works, PNC</td>
<td>Kamisaibara-mura, Okayama</td>
<td>Oct. '85</td>
<td>Less than 5%</td>
<td>200 ton SWU/year</td>
<td>Uranium enrichment</td>
</tr>
<tr>
<td>Tokai Works, PNC</td>
<td>Tokai-mura, Naka-gun, Ibaraki</td>
<td>Oct. '86</td>
<td>Less than 1.6%</td>
<td>36 ton Pu U/year</td>
<td>Fuel rod for ATR</td>
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<td></td>
<td>Plutonium;</td>
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<td></td>
<td></td>
<td>3.6X or less</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Rokkasho Works, Japan Nuclear Fuel Limited</td>
<td>Rokkasho-mura, Kamikita-gun, Aomori</td>
<td>Aug. '88</td>
<td>Less than 5%</td>
<td>600 ton SWU/year</td>
<td>Uranium enrichment</td>
</tr>
</tbody>
</table>

* The maximum capacity of chemical processing facilities includes the capacity by the conversion process, 450 ton-U/year.
### Facilities Using Nuclear Fuel Materials*  
*(As of June 30, 1993)*

<table>
<thead>
<tr>
<th>Name of Establishments</th>
<th>Location</th>
<th>Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>JAERI</strong></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
| - Tokai Research Establishment | Naka-gun, Ibaraki | - Post-irradiation tests; etc.  
- Hot lab.  
- Reprocessing research laboratory  
- No. 1 Plutonium laboratory  
- No. 2 Plutonium laboratory  
- Reactor Fuel Examination facility  
- Radioisotope production lab.  
- JRR-2  
- JRR-3  
- Waste safety testing facility  
- Waste treatment Facility  
- NSRR  
- Back-end cycle research facility | - Research on technological development of reprocessing; etc.  
- Radiochemical, solid chemical and analytical chemistry research on plutonium fuels; etc.  
- Research on compounds and property of uranium and plutonium fluorides; etc.  
- Various tests and inspections related to spent fuel; etc.  
- Separation process for radioisotope from spent fuel; technological development; etc.  
- Irradiation tests on nuclear fuel materials; etc.  
- Irradiation tests on nuclear fuel materials; etc.  
- Safety research on processing and disposal of high-level radioactive waste  
- Processing, disposal and storage of radioactive waste  
- Research on fuel rupture behavior  
- Research and development of safety control of high-level reprocessing and TRU wastes |         |
| o O'arai Research Establishment | Higashi-Ibaraki-gun, Ibaraki | - Post-irradiation test;  
- Irradiation test; mock-up test for irradiation; etc.  |

* Provided by the Government Ordinance Article 16-2.
<table>
<thead>
<tr>
<th>Name of Establishments</th>
<th>Location</th>
<th>Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td>* Plutonium fuel research facility</td>
<td></td>
<td>Research and development of new type fuels for fast breeder reactor and technology of plutonium treatment. Processing, disposal and storage of waste</td>
</tr>
<tr>
<td>* Waste treatment facility</td>
<td></td>
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<tr>
<td><strong>PNC</strong></td>
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</tr>
<tr>
<td>o Tokai Works</td>
<td>Naka-gun, Ibaraki</td>
<td>Development of fabrication technology for nuclear fuels containing plutonium and related tests; etc. Fabrication of fuels for fast experimental reactor and a new type conversion reactor; etc. Development of fabrication technology for fast breeder reactor fuel and related technologies Experiments on reduction-process technology of plutonium waste Technological development concerning reprocessing of spent fuel; etc. Technological research concerning advanced type reactor fuel reprocessing and high-level radioactive waste treatment and disposal Technological development concerning uranium enrichment Technological development concerning uranium enrichment Washing of uniforms, etc. Storage of uranium compounds Uranium wastes disposal</td>
</tr>
<tr>
<td>* Plutonium fuel development facility (PFDF)</td>
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<tr>
<td>* Plutonium fuel fabrication facility (PFFF)</td>
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<tr>
<td>* Plutonium fuel production facility (PFFF)</td>
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<td></td>
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<tr>
<td>* Plutonium waste treatment facility (PWTF)</td>
<td></td>
<td></td>
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<tr>
<td>* B Building</td>
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<td></td>
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<tr>
<td>* Chemical processing facility (CPF)</td>
<td></td>
<td></td>
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<tr>
<td>* J Building</td>
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<td>* M Building</td>
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<tr>
<td>Laundry</td>
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<tr>
<td>* Uranium storage facility-2</td>
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<td>* Uranium waste treatment facility</td>
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<tr>
<td>Name of Establishments</td>
<td>Location</td>
<td>Purpose</td>
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<td>-------------------------------------------------------------------------</td>
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<tr>
<td>O'arai Engineering Center</td>
<td>Higashi-Ibaraki-gun, Ibaraki</td>
<td>Post-irradiation test on plutonium fuel; etc.</td>
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<td>Dismantling inspection on irradiated fuel assembly, etc.; sample preparation; etc.</td>
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<td>Material test on irradiated fuel cladding, etc.</td>
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### 8. Regulatory Guides for Licensing of Nuclear Fuel Facilities

<table>
<thead>
<tr>
<th>Name of guides</th>
<th>Established date</th>
<th>Contents of guides</th>
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| "Basic guide for licensing of Nuclear Fuel Facilities" | Established by Nuclear Safety Commission on February 7, 1980; revised partially on March 27, 1989. | This guide outlines the essential concepts necessary for safety examinations generally applied to nuclear fuel facilities with the objective of unifying safety assessment. This guide defines the following five items that should be taken into account during the safety examination of nuclear fuel facilities:  
1. conditions siting  
2. radiation control  
3. environmental safety  
4. critical safety  
5. other safety measures |
| "Regulatory Guide for Licensing of Uranium Fabrication Facilities" | Established by Nuclear Safety Commission on December 22, 1980; revised partially on March 27, 1989. | This guide outlines the important fundamental items necessary for an safety examination generally applied to the facilities where uranium at an enrichment ratio less than 5% is fabricated. The objective of the guide is to achieve a rational safety examination for uranium fabrication facilities. |
| "Reference Data for Plutonium Intake in Relation to Site Evaluation of Nuclear Fuel Facilities" | Established by Nuclear Safety Commission on May 26, 1983; revised partially on March 27, 1989. | This guide defines the plutonium dose as a mark used for determining the appropriateness of the siting of a nuclear fuel facility in terms of a hypothetical accident. A plutonium dose as a mark is as following:  
1. Bone dose as a mark should be 2.4 Sv as a dose in tissues close to the bone surface.  
2. Lung dose as a mark should be 3 Sv.  
3. Liver dose as a mark should be 5 Sv. |
| "Regulatory Guide for Licensing of Reprocessing Facilities" | Established by Nuclear Safety Commission on February 20, 1987; revised partially on March 27, 1989. | This guide outlines the important fundamental items necessary for an safety examination applied to the reprocessing facilities where spent fuel from LWR power reactors is processed by the wet process (perox process). It is based fundamentally on the concept expressed in the "Basic Guide for Licensing of Nuclear Fuel Facilities" and also takes into account the characteristic features of reprocessing facilities. The items include the conditions siting an evaluation of the dose equivalent during routine operation, a safety assessment, containment function, critical safety, seismic structure, and failures. |
9. Dose Equivalents to Workers in Nuclear Facilities in Fiscal 1992

1. The owners and the users of refining facilities, processing facilities, and reprocessing facilities are required to control dose equivalents to the persons working in these facilities so that dose equivalents will not exceed the dose equivalent limits which are defined by the Notification for the Law for the Regulation on Nuclear Source Materials, Nuclear Fuel Materials, and Nuclear Reactors.

The data mentioned below have been prepared from the reports on radiation management in fiscal 1991 submitted by owners and users of refining facilities, processing facilities, and reprocessing facilities (limited to those owners and users of facilities for nuclear fuel materials as defined by Article 16-2 of the Government Ordinance) in accordance with the Law for the Regulation on Nuclear Source Materials, Nuclear Fuel Materials, and Nuclear Reactors, concerning effective dose equivalents to the workers in these facilities. The data have been prepared also from the reports on radiation management submitted in accordance with the administrative notification.

These data show that effective dose equivalents to workers in refining facilities and processing facilities as well as reprocessing facilities did not exceed the dose equivalent limits in fiscal 1991. The tissue dose equivalents to the two workers, however, exceeded the dose equivalent limits at the time they were exposed to radiation on January 9, 1992, due to the trouble in the High-level Radioactive Substances Research Institute, the Power Reactor and Nuclear Fuel Development Corporation Tokai Works.

2. Definitions of Terms in the Tables

(1) “Employees” refer to the personnel and researchers who are employed in the facilities concerned, while “non-employees” refer to contractors, etc.

(2) Regarding dose equivalents, total dose equivalents have been calculated by rounding them at one decimal place under the classifications of employees, etc. The difference between the sum of the dose equivalent to the employees and that to the non-employees in some cases should be regarded as an error resulting from this method of calculation.

(3) The “mean dose equivalent” has been found by rounding it to one decimal place.

(4) The meanings of “0” and “0.0” in these Tables are as follows:

1  “0” in the total dose equivalent column: 0 or less than 0.5
2  “0” in the mean dose equivalent column: 0
   “0.0” : less than 0.05
### Distribution of dose equivalents

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<th>Exceeding 10 mSv, 15 mSv or less</th>
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<th>Sub-total</th>
<th>Exceeding 25 mSv, 50 mSv or less</th>
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<th>Total dose equivalent (persons x mSv)</th>
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**Dose Equivalents to Workers in Nuclear Facilities in Fiscal 1992**

(April 1992 through March 1993)
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<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>774</td>
<td></td>
<td>774</td>
<td>98</td>
<td>0.0</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

175
<table>
<thead>
<tr>
<th>Establishment</th>
<th>Employees</th>
<th>Non-employees</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>5. JAERI Narai Institute (JNIR Hot Lab., etc.)</td>
<td>103</td>
<td>277</td>
<td>380</td>
</tr>
<tr>
<td>6. Material Testing Reactor Facilities attached to the Metal Materials Research Institute of Tohoku University</td>
<td>20</td>
<td>14</td>
<td>34</td>
</tr>
<tr>
<td>7. Nippon University Engineering Faculty (Hot Lab. Facilities)</td>
<td>94</td>
<td>0</td>
<td>94</td>
</tr>
<tr>
<td>8. High-energy Physics Research Institute (Neutron Experiment Facilities)</td>
<td>30</td>
<td>0</td>
<td>30</td>
</tr>
<tr>
<td>9. National Institute of Radiological Medicine (Internal Exposure Experiment Building)</td>
<td>48</td>
<td>127</td>
<td>175</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Distribution of dose equivalents</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exceeding 15 mSv, 25 mSv or less</td>
</tr>
<tr>
<td>Exceeding 15 mSv, 20 mSv or less</td>
</tr>
<tr>
<td>Exceeding 25 mSv, 50 mSv or less</td>
</tr>
<tr>
<td>Total dose equivalent (pers- x mSv)</td>
</tr>
<tr>
<td>----------------------------------</td>
</tr>
<tr>
<td>105</td>
</tr>
<tr>
<td>258</td>
</tr>
<tr>
<td>363</td>
</tr>
<tr>
<td>20</td>
</tr>
<tr>
<td>14</td>
</tr>
<tr>
<td>34</td>
</tr>
<tr>
<td>94</td>
</tr>
<tr>
<td>94</td>
</tr>
<tr>
<td>30</td>
</tr>
<tr>
<td>30</td>
</tr>
<tr>
<td>41</td>
</tr>
<tr>
<td>117</td>
</tr>
<tr>
<td>158</td>
</tr>
<tr>
<td>Classification</td>
</tr>
<tr>
<td>----------------</td>
</tr>
<tr>
<td>Establishment</td>
</tr>
<tr>
<td>10. Nuclear Materials Management Center's Laboratory for Security Measure Analysis (Development and Examination Building)</td>
</tr>
<tr>
<td>Employees</td>
</tr>
<tr>
<td>Non-employees</td>
</tr>
<tr>
<td>Total</td>
</tr>
<tr>
<td>11. Nuclear Fuel Industry Total Works'3</td>
</tr>
<tr>
<td>Employees</td>
</tr>
<tr>
<td>Non-employees</td>
</tr>
<tr>
<td>Total</td>
</tr>
<tr>
<td>12. Japan Nuclear Fuel Development Co. (JFD Hot Lab. Facilities)</td>
</tr>
<tr>
<td>Employees</td>
</tr>
<tr>
<td>Non-employees</td>
</tr>
<tr>
<td>Total</td>
</tr>
<tr>
<td>Employees</td>
</tr>
<tr>
<td>Non-employees</td>
</tr>
<tr>
<td>Total</td>
</tr>
<tr>
<td>14. Mitsubishi Material Co. Maka Nuclear Power Development Center (Second Development and Examination Building)</td>
</tr>
<tr>
<td>Employees</td>
</tr>
<tr>
<td>Non-employees</td>
</tr>
<tr>
<td>Total</td>
</tr>
</tbody>
</table>
### Distribution of dose equivalents

<table>
<thead>
<tr>
<th>Classification</th>
<th>5 mSv or less</th>
<th>Exceeding 5 mSv, 15 mSv or less</th>
<th>Exceeding 15 mSv, 25 mSv or less</th>
<th>Exceeding 25 mSv, 50 mSv or less</th>
<th>Number of workers (persons)</th>
<th>Total dose equivalent (persons x mSv)</th>
<th>Mean dose equivalent (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Establishment</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>Employees</td>
<td>1,623</td>
<td>17</td>
<td></td>
<td></td>
<td>1,640</td>
<td>371</td>
</tr>
<tr>
<td></td>
<td>Non-employees</td>
<td>3,896</td>
<td>61</td>
<td></td>
<td></td>
<td>3,957</td>
<td>1,036</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>5,519</td>
<td>78</td>
<td></td>
<td></td>
<td>5,597</td>
<td>1,407</td>
</tr>
<tr>
<td>Grand total</td>
<td>Employees</td>
<td>3,796</td>
<td>18</td>
<td></td>
<td></td>
<td>3,816</td>
<td>1,014</td>
</tr>
<tr>
<td></td>
<td>Non-employees</td>
<td>6,925</td>
<td>69</td>
<td></td>
<td></td>
<td>6,994</td>
<td>1,680</td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>10,723</td>
<td>87</td>
<td></td>
<td></td>
<td>10,810</td>
<td>2,694</td>
</tr>
</tbody>
</table>


*2: Nuclear Fuel Industry, which had been engaged in processing, fell under the classification as nuclear fuel material using facilities which are defined by Article 16-1 of the Government Ordinance. As the number of workers includes that of workers engaged in processing, the latter number has been excluded from the total.
10. Conditions of Radioactive Waste Management in Reprocessing Facilities

(1) In the reprocessing facilities of the Tokai Works of the Power Reactor and Nuclear Fuel Development Corporation, radioactive waste gases and liquids are controlled so that they will not exceed the discharge limits defined by the security regulations applied in the facilities. Concentrated high-level radioactive waste liquids are storage-discarded in high-level radioactive waste storage tanks, while low-level radioactive waste liquids and sludge as well as spent solvents are either storage-discarded in tanks respectively or solidified with asphalt or plastic.

High-level radioactive waste solids are conveyed in casks which are used exclusively for this purpose to the high-level radioactive solid waste storehouse or to the second storage facilities for high-level radioactive solid wastes. These wastes are stored in water or dry-stored under the prescribed methods. Low-level radioactive waste solids are sealed in drums, etc., and stored in the first or the second storage facilities for low-level radioactive solid wastes. Wastes which have been solidified with asphalt or plastic are sealed in drums and storage-discarded in the storage facilities for asphalt-solidified radioactive wastes or the second facilities for asphalt-solidified radioactive wastes.

(2) The data in this report have been prepared from the report on radiation management for fiscal 1991 which has been submitted by the Power Reactor and Nuclear Fuel Development Corporation, a nuclear fuel reprocessor, in accordance with the Law for the Regulation on Nuclear Source Materials, Nuclear Fuel Materials, and Nuclear Reactors.

The amounts of radioactive gaseous wastes and radioactive liquid wastes discharged from the Corporation were both below the discharge limits.

The amount of radioactivity of concentrated low-level waste liquids became smaller than that of the previous year due to the attenuation of their radioactivity.
(3) The quantity of spent fuel which underwent reprocessing was 81.7 tons in fiscal 1991.

(4) The meanings of terms and symbols in the tables which show conditions of radioactive waste management in the reprocessing facilities are mentioned below.

1 Radioactivity of radioactive waste gases and radioactive waste liquids discharged from the facilities has been measured with the total alpha radioactivity measuring method, the total beta radioactivity measuring method, liquid scintillation method, gamma-ray spectrometry, etc.

2 The quantity of high-level solid wastes shows the amount of high-level solid wastes which have been conveyed in casks to be used exclusively for this purpose. The quantities of low-level radioactive solid wastes and asphalt-solidified wastes as well as plastic-solidified wastes show the amounts sealed in drums and stored. The amount of large wastes which cannot be sealed in drums is shown as being sealed in 200-liter drums.

3 "N.D." in the tables refers to levels below detection limits.
1. Amount of Radioactive Wastes Discharged during Fiscal 1992

(1) Gaseous radioactive wastes

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Limit of annual discharge (GBq)</th>
<th>Annual amount of discharge (GBq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{85}$Kr</td>
<td>$8.9 \times 10^7$</td>
<td>$9.8 \times 10^6$</td>
</tr>
<tr>
<td>$^3$H</td>
<td>$3.6 \times 10^3$</td>
<td>$2.8 \times 10^3$</td>
</tr>
<tr>
<td>$^{14}$C</td>
<td>$9.7 \times 10^3$</td>
<td>$7.8 \times 10^2$</td>
</tr>
<tr>
<td>$^{129}$I</td>
<td>1.7</td>
<td>$3.0 \times 10^{-1}$</td>
</tr>
<tr>
<td>$^{131}$I</td>
<td>$1.6 \times 10$</td>
<td>N.D. (Note 1)</td>
</tr>
</tbody>
</table>

Note 1: Detection limit: $3.7 \times 10^{18}$ Bq/cm$^3$
(2) Liquid radioactive wastes

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Limit of annual discharge (GBq)</th>
<th>Annual amount of discharge (GBq)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total alpha radioactivity</td>
<td>4.1</td>
<td>$7.4 \times 10^{-3}$</td>
</tr>
<tr>
<td>Total beta radioactivity (excluding $^3$H)</td>
<td>$9.6 \times 10^2$</td>
<td>N.D. (Note 1)</td>
</tr>
<tr>
<td>$^{39}$Sr</td>
<td>$1.6 \times 10^1$</td>
<td>N.D. (Note 2)</td>
</tr>
<tr>
<td>$^{90}$Sr</td>
<td>$3.2 \times 10^1$</td>
<td>N.D. (Note 3)</td>
</tr>
<tr>
<td>$^{95}$Zr-$^{95}$Nb</td>
<td>$4.1 \times 10^1$</td>
<td>N.D. (Note 4)</td>
</tr>
<tr>
<td>$^{103}$Ru</td>
<td>$6.4 \times 10^1$</td>
<td>N.D. (Note 5)</td>
</tr>
<tr>
<td>$^{95}$Zr-$^{106}$Gd</td>
<td>$5.1 \times 10^2$</td>
<td>N.D. (Note 6)</td>
</tr>
<tr>
<td>$^{134}$Cs</td>
<td>$6.0 \times 10^1$</td>
<td>N.D. (Note 7)</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>$5.5 \times 10^1$</td>
<td>$2.5 \times 10^{-2}$</td>
</tr>
<tr>
<td>$^{141}$Ce</td>
<td>$5.9$</td>
<td>N.D. (Note 8)</td>
</tr>
<tr>
<td>$^{144}$Ce-$^{144}$Pr</td>
<td>$1.2 \times 10^2$</td>
<td>N.D. (Note 9)</td>
</tr>
<tr>
<td>$^3$H</td>
<td>$1.9 \times 10^4$</td>
<td>$3.8 \times 10^3$</td>
</tr>
<tr>
<td>$^{129}$I</td>
<td>$2.7 \times 10^1$</td>
<td>$6.5 \times 10^{-3}$</td>
</tr>
<tr>
<td>$^{131}$I</td>
<td>$1.2 \times 10^2$</td>
<td>N.D. (Note 10)</td>
</tr>
<tr>
<td>Pu (α)</td>
<td>2.3</td>
<td>$4.0 \times 10^{-3}$</td>
</tr>
</tbody>
</table>

Notes:
1. Detection limit $2.2 \times 10^{-2}$ Bq/cm$^3$
2. Detection limit $2.2 \times 10^{-2}$ Bq/cm$^3$
3. Detection limit $1.1 \times 10^{-3}$ Bq/cm$^3$
4. Detection limit $4.3 \times 10^{-3}$ Bq/cm$^3$
5. Detection limit $1.1 \times 10^{-3}$ Bq/cm$^3$
6. Detection limit $3.2 \times 10^{-2}$ Bq/cm$^3$
7. Detection limit $1.1 \times 10^{-3}$ Bq/cm$^3$
8. Detection limit $2.2 \times 10^{-2}$ Bq/cm$^3$
9. Detection limit $2.2 \times 10^{-2}$ Bq/cm$^3$
10. Detection limit $1.8 \times 10^{-2}$ Bq/cm$^3$
2. Amount of Radioactive Waste in Storage

(1) Radioactive waste liquids in storage

<table>
<thead>
<tr>
<th>Classification</th>
<th>Concentrated high-level radioactive waste liquids</th>
<th>Concentrated low-level radioactive waste liquids</th>
<th>Sludge</th>
<th>Spent solvents</th>
</tr>
</thead>
<tbody>
<tr>
<td>Amount in storage as of the end of March 1993 (m³)</td>
<td>516</td>
<td>1,314</td>
<td>1,088</td>
<td>38</td>
</tr>
<tr>
<td>Amount of radioactivity as of the end of March 1993 (beta GBq)</td>
<td>$4.6 \times 10^5$</td>
<td>$3.9 \times 10^5$</td>
<td>$1.4 \times 10$</td>
<td>$2.7 \times 10^4$</td>
</tr>
</tbody>
</table>

(2) Amounts of radioactive waste solids created and stored

<table>
<thead>
<tr>
<th>Classification (Classification of storage containers)</th>
<th>High-level radioactive waste solids</th>
<th>Low-level radioactive waste solids</th>
<th>Asphalt-solidified wastes</th>
<th>Plastic-solidified wastes</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sheared covering sections (hull, etc.) (350-liter can)</td>
<td>Spent clarifying filters</td>
<td>Sample bottles, etc. (20-liter can)</td>
<td>Number of drums (200-liter drum)</td>
<td>Number of drums (200-liter drum)</td>
</tr>
<tr>
<td>Amount created in FY 1992</td>
<td>101</td>
<td>11</td>
<td>189</td>
<td>1,118</td>
</tr>
<tr>
<td>Amount stored as of the end of March 1993</td>
<td>1,076</td>
<td>87</td>
<td>3,245</td>
<td>22,665</td>
</tr>
</tbody>
</table>
(Reference data) Actual Amounts of Radioactive Wastes

### (1) Radioactive waste gases

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Cresylic acid</td>
<td>1.9x10^-4</td>
<td>2.0x10^-4</td>
<td>2.3x10^-4</td>
<td>2.6x10^-4</td>
<td>2.8x10^-4</td>
<td>2.6x10^-4</td>
<td>2.4x10^-4</td>
<td>2.2x10^-4</td>
<td>2.1x10^-4</td>
<td>2.0x10^-4</td>
<td>1.8x10^-4</td>
<td>1.7x10^-4</td>
<td>1.5x10^-4</td>
</tr>
<tr>
<td>Tritium</td>
<td>1.7x10^-6</td>
<td>1.9x10^-6</td>
<td>2.0x10^-6</td>
<td>1.7x10^-6</td>
<td>1.5x10^-6</td>
<td>1.4x10^-6</td>
<td>1.3x10^-6</td>
<td>1.2x10^-6</td>
<td>1.0x10^-6</td>
<td>8.0x10^-7</td>
<td>6.0x10^-7</td>
<td>4.0x10^-7</td>
<td>3.0x10^-7</td>
</tr>
<tr>
<td>Carbon dioxide</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
<td>5.7x10^-4</td>
</tr>
</tbody>
</table>

**Notes:**
1. The management system was changed in September 1988 from the elemental management system to the mobile management system.
2. Amounts have been expressed in kg since April 1989.
3. Detection limit: 7 x 10^-6 kg/m^3 (9 x 10^-6 kg/m^3 for CH4 gases)
4. Values due to leakage of the...........................................................................
### Outline of Rokkasho Nuclear Fuel Cycle Facilities

**(As of June 1993)**

<table>
<thead>
<tr>
<th>Undertaker</th>
<th>Reprocessing Plant</th>
<th>Uranium Enrichment Plant</th>
<th>Low-level Radioactive Waste Burial Center</th>
</tr>
</thead>
<tbody>
<tr>
<td>Way</td>
<td>Reprocessing capacity: 800 tU per year</td>
<td>Glass-solidified high-level waste storage capacity: 1,640 pieces for the time being</td>
<td>To be increased by 150 tSWU/year from 1991 so that the capacity will reach the level of 1,500 tSWU/year. 1,500 tSWU is equivalent to the capacity to produce the necessary amount of enriched uranium for re-placed fuel from more than 10 power reactors of 1,000,000 kW class. Already approved amount: 600 tSWU/year</td>
</tr>
<tr>
<td></td>
<td>Capable of reprocessing replaced fuel from approx. 30 power reactors of 1,000,000 kW class. Spent fuel storage capacity: 3,000 tU</td>
<td>(It is to be increased to the level of 3,000 pieces.) Glass-solidified high-level wastes treated here refer to those wastes which have been returned after reprocessing abroad.</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Scale of facilities</th>
<th>Reprocessing Plant</th>
<th>Japan Nuclear Fuel Service Co.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Approx. 3,600,000 m² (including green belts and roads)</td>
<td>Approx. 3,600,000 m² (including green belts and roads)</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Area of plant site</th>
<th>Approx. 3,600,000 m² (including green belts and roads)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Approx. 3,600,000 m² (including green belts and roads)</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Schedule</th>
<th>Application for designation: March 1989</th>
<th>Application for approval: March 1989</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety examination has been finished in the Science and Technology Agency, and the application is being examined by both the Atomic Energy Commission and the Nuclear Safety Commission. Operations are planned to be started in 1999.</td>
<td>Approved: April 1992</td>
<td>Start of operations: March 1992</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Services by legal classification</th>
<th>Reprocessing</th>
<th>Waste management</th>
<th>Processing</th>
<th>Waste burial</th>
</tr>
</thead>
</table>

186
The UK Nuclear Regulatory Structure as Applied to Chemical Engineering Assessment in the Nuclear Fuel Cycle

Presented by:

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Presented to:
A Topical Meeting on Fuel Cycle Safety
OECD NEA
Cadarache
France

September 1994
SUMMARY

In the first part of this paper, I briefly outline the UK regulatory regime following the introduction of the "Standard Licence" in late 1990. I then discuss the philosophy developed in HSE's publication Tolerability of Risk from Nuclear Power Stations and how it is applied in our Safety Assessment Principles for Nuclear Plants. I show how the Inspectorate is organised to carry out this work and I address the relationship between Probabilistic Safety Analysis and the underlying engineering principles. Finally I draw out the relationship between the Law, the Licence and our guidance in both the general sense and as it is applied to the nuclear fuel cycle safety assessment.

In the second part of this paper, I discuss the application of a few Key Principles to show that it is in the interests of both the regulator and the industry to exchange current conceptual thinking about plant and processes before the design work has significantly progressed. I then relate this directly to the Chemical Engineering aspects and draw out examples where such an approach has proved both useful and fruitful for both parties. I then draw out the lesson that a simple well organised liaison on a regular basis provides dividends for both parties although it is not specifically covered under the regulatory controls embodied in the licence or other legislation.
INTRODUCTION

1. Since the beginning of the civil nuclear power programme in the UK the industry has been regulated under the Nuclear Installations Act (NIA). In 1974, the safety aspects of the NIA, with its associated licensing regime, came under the Health and Safety at Work etc Act 1974 (HSWA). This Act 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). Under the NIA no site may be used for the purposes of installing or operating any prescribed nuclear installation unless a nuclear site licence has been granted by the HSE and is, for the time being, in force. HM Nuclear Installations Inspectorate (NII) is that part of HSE responsible for administering this licensing function.

2. The HSWA places a general duty on employers to ensure, so far as is reasonably practical, the health safety and welfare of all their employees and also to conduct their undertaking in such a way as to ensure, so far as is reasonably practical, that people not in their employment who may be affected are not thereby exposed to risk to their health or safety. Specific requirements for the radiological protection of employees and the public are contained in the Ionising Radiations Regulations 1985 made under the HSWA. Other specific nuclear safety requirements are a matter for NII acting for HSE to formulate and apply. These requirements are usually in the form of conditions attached to the site licence. Licenses are granted for specific sites and they can be tailored to each site. In practice there are standard conditions attached to licenses for all major licensees. The licensees' response to these requirements may take different forms; for example, as a safety case relating to a stage in the life of an installation or as arrangements or procedures to meet the requirements of a licence condition. Additionally, guidance is set out in our Safety Assessment Principles [1] (SAPs - see paragraph 16) which we have developed for our own use in judging the adequacy of licensees safety cases. We have further guidance in the recently republished Notes for Applicants [2]. In all UK safety legislation and the material which flows from it there is the concept of controlling risks to a level which will be As Low as Reasonably Practical (ALARP - see paragraph 15).

3. The duty of NII 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. In all this, the licensee remains solely responsible for safety.

THE STANDARD LICENCE

4. The heart of the regulatory control system 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. 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).

5. Thus, every licence is unique in detail to its site and it may be varied to exclude any part of the site the licensee no longer requires for licensable activities. Before making such a variation the NII, 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.

6. Similarly, a licence may be revoked by the NII or surrendered by the licensee. In either event, the licensee will remain responsible for the safety of activities on that site. This period of responsibility ends when a new licence has been issued or NII gives written notice that, in its opinion, there has ceased to be any danger from ionising radiations from anything on that site.

7. The NII 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. However, a standard set of licence conditions has evolved with the aim of producing consistent safety requirements which are non-prescriptive and flexible. These are shown in Appendix I. Among other things they deal with:

   a. marking the site boundary
   b. appointment of "Duly Authorised Persons"
   c. the production of safety cases
   d. incident reporting
   e. operating rules as 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 - see also paragraph 36).
   i. emergency arrangements.

8. In the main, these 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.

9. The licensee's arrangements are elements of an overall safety management system. NII reviews these arrangements to ensure they are clear, unambiguous and adequately address the main safety matters. In particular, NII 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

NII ORGANISATION

10. HSE's mission is to ensure that risks to people's health and safety from work activities are controlled properly. HSE consists of 11 divisions (see figure 1) of which Nuclear Safety Division is one. NII is the major part of Nuclear Safety Division and the divisional structure is shown in figure 2. The six branches are organised in a system of matrix management. The two policy branches provide such things as day to day help and information to the Secretary of State for Trade and Industry and his principal advisers, advice to the Health and Safety Commission, nuclear licensing policy, co-ordination of NII's international work and formulation of radiation protection policy for HSE. These Branches also provide management functions for NII's external research support, training management, and other management services including secretariat to a number of advisory committees.

11. The two inspection branches work on a project basis and administer the site licenses for nuclear power stations and other nuclear sites. Important features of their work include monitoring plant modifications and emergency exercises.

12. The two assessment branches support the inspection and policy branches providing assessments, usually of licensees' safety cases. The topics are also shown in figure 2. Broadly, Branch B is assigned analytical topics and Branch C engineering, although that is an over simplification.

13. In this multi-faceted structure the overall aim can be summarised as:

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.

The structure gives the following characteristics to achieve this aim:
a. the ability to decide priorities centrally, across all branches
b. the ability to respond rapidly to technical innovation and operational experience
c. more consistency in standards as they are developed and applied
d. clarification of responsibilities for each project
e. coherent and consistent regulatory requirements as they are developed and applied
f. advantages in personal development to widen experience and enrich inspectors' careers.

NII, as part of NSD and HSE is well organised to meet the demands likely to be placed upon it.

TOLERABILITY OF RISK

14. HSE's document on the Tolerability of Risk (TOR) originates from a recommendation by Sir Frank Layfield in his 1986 report of the Sizewell Inquiry [8] into the UK's first PWR. This recommended that HSE should formulate and publish guidelines on the tolerable levels of individual and societal risk to workers and the public from nuclear power stations. He said the opinion of the public should underlie the evaluation of risk but there was then insufficient public information to allow understanding of the basis for the regulation of nuclear safety. He recommended that as a first step, HSE should publish a document aimed at initiating public, expert and parliamentary debate.

15. In response to Sir Frank Layfield's recommendation HSE produced the first version of TOR in 1988. Public comment was invited and, following close public scrutiny at the public inquiry to build UK's second PWR at Hinkley Point TOR was republished [3]. TOR is a straightforward account written for the general public. It discusses how people normally approach risk, shows how industrial risks (and nuclear risks in particular) are regulated, the nature of risk from radiation and how these are calculated. In doing this it established three levels of risk:

a. a risk which is so great or the outcome so unacceptable that it must be refused altogether - which can be shortened to unacceptable or intolerable risks: these cannot be justified except in extraordinary circumstances

b. a risk which is or has been made so small that no further precaution is necessary - the "broadly acceptable" region where no detailed working is needed to show that risks are ALARP

c. risks that fall between these two states, that have been reduced to the lowest level practicable bearing in mind the benefits flowing from their acceptance and taking into account the cost of further risk reduction. The injunction laid down in safety law is that any such risk must be reduced so far as is reasonably practical. This is the ALARP or Tolerability region

TOR goes on to quantify these regions for individual risks. This, in turn, forms the basis for some of the quantitative limits found in SAPs.
SAFETY ASSESSMENT PRINCIPLES FOR NUCLEAR PLANT

16. The current SAPs [1] apply to all prescribed civil nuclear installations and supersede separate documents developed for reactors [4] and nuclear chemical plant [5]. It is important to note that the SAPs are at the "Principle" level and are not generally targeted at any particular type of installation. They are produced to guide inspectors on the assessment of safety cases and to promote consistent standards. They are not standards imposed on licensees but they have been published so that anyone who is interested can be aware of the safety criteria against which licensees' safety cases will be judged. In common with the licence they are non prescriptive and are intended for use with new plant and major plant modifications. However, they are also used in safety reviews of older plant – which are required under licence conditions – for comparison with modern standards and to give a benchmark against which any argument on what is reasonably practical can be set. SAPs are not intended as an operator's design document although there is a need to minimise avoidable inconsistencies.

17. The bulk of the SAPs set out NII's views on good engineering practice which can be regarded as the basis of safe design. However, the overall risk targets incorporated in the probabilistic analysis to achieve a balanced design derive their numerical criteria from TOR. The doses from normal operation and those from accidents have both a limit of tolerability denoted by a Basic Safety Limit (BSL) and a broadly acceptable level or Basic Safety Objective (BSO). For example, accidents which lead to off site doses to members of the public are expressed in terms of dose frequency relationships which reflect the well accepted concept of decreasing frequency with increasing dose.

18. It is intentional that the bulk of the SAPs address engineering aspects. It is only by matching the quality of the engineering to the safety significance of the potential faults that the risk targets can be demonstrably met. In other words, without the quality of engineering underlying the fault analysis there is a considerable difficulty in engendering confidence in the risk based numbers. It is at this point that the deterministic meets the probabilistic and such aspects as engineering standards, quality research and development and engineering judgement play an important part.

19. The SAPs do make specific reference to chemical plant in a number of places (although the number is small), particularly when interpretation of a principle is required. The most obvious is in the explanation of Principle P45 which states:

"Such plant damage is interpreted as a degraded core in the case of a reactor. For other plant, it would include a major breach of a vessel pipework etc., together with the potential for events such as fire, explosion or aggressive chemical attack which might lead to degradation of the containing cell or its ventilation/filtration system even though there may be a safety system provided to prevent such degradation".

THE RELATIONSHIP BETWEEN THE LAW AND OTHER COMPONENTS OF THE REGULATORY FRAMEWORK

20. A recent review of UK Health and Safety Regulation [6] contained a number of recommendations. One of these was:
"the system of health and safety regulation includes legislation, Approved Codes of Practice and guidance. The respective roles of these components are not well understood. Non-mandatory guidance is widely - but wrongly - seen as imposing obligations on business, and the limited role of Approved Codes of Practice (ACOPs) is not generally appreciated."

Whilst this is not particularly relevant to our licensees who have a close working relationship with NII, it is not always the case for small companies. Also, there are good reasons to clarify the relationship between these in the broadest sense since it improves general understanding of the UK regulatory process and the nuclear regulatory process in particular.

21. The HSWA, its relevant statutory provisions (of which the NIA is one) and regulations made under section 15 of the HSWA together constitute the legislative framework which is applied to civil nuclear installations. Compliance is mandatory and contravention is a matter for appropriate regulatory action. Regulations have the same force of law as the HSWA and contraventions are treated in the same way. The NIA empowers HSE to attach conditions to a licence at any time which appear necessary or desirable in the interest of safety. This makes the licence conditions equivalent to Regulations.

22. The section 16 of HSWA also makes provision for "Approval of Codes of Practice":

"For the purposes of providing practical guidance with respect to the requirements of any of the provisions (given in the section) or of any existing statutory provisions ... ."

The important aspect is practical guidance. Such ACOPs do not constitute the law but provide help by showing ways to comply with the law. Compliance with an ACOP is a defence in law but non-compliance is not an offence provided equivalent safety can be demonstrated by other means. This is consistent with the non prescriptive concept of the UK safety regime. Of particular relevance is the ACOP associated with the Ionising Radiations Regulations (IRRs). This provides general guidance particularly relevant to small companies who may not have the "in house" expertise available to our major licensees.

23. HSE also issues general guidance to be helpful to those affected by the legislation as empowered by Section 11 of HSWA. Such guidance has no statutory standing in the way an ACOP has but it may be used in a court of law as evidence illustrating good industrial practice. HSE's recently published the Notes for Applicants [2] is this type of guidance. Conversely, TOR and SAPs show how NII judge compliance with the licence and other duties under HSWA such as the IRRs. As with guidance, they have no statutory basis but may be used in public inquiries, for example.

24. This Part has covered the overall framework for safety regulation in the UK. In such a short paper it is impossible to cover all the relevant aspects but it does make clear the relationships between the components of the regulatory regime. The crucial point is that NII does not prescribe in detail how the licensees should comply with their legal obligations. It is for licensees to present their safety cases as they see fit and for NII to judge them against the SAPs which are high level goals. There is considerable scope for flexibility and judgement by both the licensee and the regulator.
PART II

INTRODUCTION

25. The regulatory regime of licensing, regulations and guidance has been outlined in Part I and the underlying relationships have been established. This Part demonstrates how a specialist Chemical Engineer can be involved in the assessment of safety cases and some of the significant features we might look for with reference to a number of SAPs in the predominantly non prescriptive regulatory regime of licensing, regulations and guidance in the UK. Whilst most, if not all, of the SAPs apply to chemical plant as they do to power reactors, I will concentrate on certain SAPs with prime engineering importance.

ASSESSMENT

26. In this non prescriptive regime the licensee is free to propose any means to achieve an appropriate level of safety. However, he must demonstrate in a clear and unambiguous fashion that his proposals are adequate. This demonstration must be objective and traceable. Traceability or transparency usually means that it is auditable and documented. Thus there should be a clear link between the underlying concept, the supporting research and development and the design. The design, in turn, should be clearly linked to standards germane to the safety duty of the item(s) to which they are applied. In all this there must be a quality regime and account must be taken of relevant industrial experience both in the UK and internationally. The resulting safety case (sometimes known as a safety justification) may vary from a few pages or equivalent, to many volumes. The size depends heavily on the complexity of the underlying technology, its safety significance and the risk from the plant to which it relates. The usual documentation stages relevant to safety in a large project before operation are [7]:

a. the reference design
b. The Preliminary Safety Report (PSR)
c. The Contract Design Report (CDR)
d. The Pre-Construction Safety Report (PCSR)

and later
e. the Pre-Commissioning Safety Report (PCmSR)
f. the Pre-Operational Safety Report (POSR)

The design reports eg the CDR, are usually only used for very large projects such as the PWR at Sizewell as sufficient information can be provided in the safety reports for smaller plant and modifications. Site inspectors also need to see the operational limits and conditions drawn out in compliance with the relevant licence conditions. The later safety reports fulfil that role.

27. However, although each safety report is tailored to a particular phase of the plant life, there are similarities in the way they are used in NII. Initially, I intend to concentrate on the similarities for major projects and safety significant modifications. It is important to note that
any assessment by the regulator does not remove the licensee's responsibility for safety. Any safety case will undergo scrutiny, including the supporting documents which form an integral part of the safety case. The NII will use its experience and expertise to sample a safety case in order to gain confidence in the arguments and how the licensee has met his own criteria taking account of their safety significance. These will be examined to ensure a complete and traceable safety argument. This assessment is the start of a probing process where the inspector carrying out the assessment (the assessor or assessment inspector) tests the licensee's claims and assumptions on the selected parts of the safety case until satisfied with the arguments. Often this can result in changes to the safety case as well as plant. However, the non prescriptive aspect still applies and the licensee is still responsible for demonstrating that plant safety is adequate both as a result of any change (safety case or plant) and during the change itself (for a physical change).

28. Thus the implementation of any particular technology is a matter for the licensee. It would be wrong for the NII to insist on the use of a given technology in any particular application. What is required at the PSR stage is that the licensee identifies and demonstrates clearly how the safety criteria cited in that report will be met; in other words, how the plant and process will fulfil its intended safety duty. The full justification will follow in the PCSR once the design has been better developed. The licensee will also have shown that the standards to be implemented eg ISO 9000, meet the safety criteria and are matched to their safety significance. For large organisations these are usually established centrally outside any particular project framework and the demonstration will be that the corporate criteria are met.

APPLICATION OF THE SAFETY ASSESSMENT PRINCIPLES

29. The safety case assessment is usually managed in NII by a project officer who may be a site inspector or a member of a specialist project management unit. This inspector will seek advice from the assessor in a variety of disciplines, relevant to that safety case. In carrying out this assessment the bench mark for most assessors are the SAPs. All of this is subject to the test of reasonable practicability and this argument is always open to the licensee.

30. The rationale underlying the qualitative principles is simple, all hazards must be avoided or adequately contained. Thus the first task for the chemical engineer is to ascertain if the process proposed by the licensee has been engineered to avoid potential risks and thus be as benign as possible. This is reflected in a number of SAPs. Most notably Key Principles:

P61 Potential hazards from operation of the proposed plant should be identified. The design concept should be such that these Hazards are avoided and safe conditions maintained through inherent and, where appropriate, passive features of the design without reliance on control or safety systems.

P62 The design concept should be such that the sensitivity of the plant to potential faults is minimised. The expected plant response to any initiation fault should be as near the top of the following list as can reasonably be achieved:

a. a failure or maloperation should produce no significant operational response, or should produce a change in the plant state towards a safer condition;
b. following a failure or maloperation, the plant should be rendered safe by the action of passive features or engineered safeguards which are continuously available in the state required to control the fault;

c. following a failure or maloperation the plant should be rendered safe by the action of active engineered safeguards which need to be brought into service in response to the fault.

and

*P64*  *For chemical plants:*

(a) strongly exothermic, or high pressure reactions should be avoided, and any source of energy released into the system should be adequately controlled as should the state of nuclear matter in the plants:

(b) the choice of process materials, their inventories, the process conditions and containment materials should be such as to minimise the consequences of potential faults, and the use or generation of hazardous or toxic materials should be avoided;

and

(c) the process flowsheet should ensure that process deviations will not move the plant towards an unsafe state

This reflects the general point made in the introduction to the Engineering Principles:

"However, some (of the principles) are of greater importance than others. They may have a major influence on the cost of the plant; or they may not have such an effect but are seen as a fundamental engineering requirement in a safe plant. They may be in neither of these categories but are nevertheless of prime importance - the need for the plant to be based on a sound concept, for example. ........"

and in the previous SAPs [4,5] which said:

"Design, manufacture, construction and operation are all key features in the safety of plant. A sound design concept, a well engineered and proven design and a high quality of manufacture and construction will be required. A high standard of organisation, ....."

31. Overall the engineering quality aspects have always been prominent in the NII approach to safety and they continue to be so. I particularly stressed the importance of the underlying concept for each plant. Whilst, for nuclear chemical plant there is little choice about the primary feedstock there are an ensuing plethora of choices for the other inputs to the process and thereby the process itself. These might be assessed by the operator using one of a number of structured approaches of which HAZOP is the best known in the UK. A HAZOP type approach can be used to assist process choices. Such processes are akin to a technical brain storming session (or sessions) guided by a structure, a number of key words in the HAZOP case, applied to the entire process to identify potential for adverse occurrences. In safety terms, the result should be a safety ranking for the candidate processes in terms of their
potential threat to individuals and the environment taken with the known means to control such threats. The technique is widely applied in the conventional chemical industry but has rarely been seen applied to rank processes in the nuclear industry. It has become common for the process choice to be included in safety cases but not usually in structured form. However, it is often apparent that some similar thinking processes have been carried out. It would enhance confidence significantly for the regulator if such information reflecting a structured and systematic approach to process selection were included at an early stage, for example, the PSR.

CONCEPTUAL DESIGN AND LIAISON

32. In the nuclear fuel reprocessing field the regulator has the benefit of the skill and experience of the chemical or process engineer. Early involvement from the chemical engineering assessor during the licensee's process selection, in liaison with the inspection Branch, will often yield dividends. In the UK regulatory regime the process choice is a matter for the licensee to justify and in doing so it must be shown that it would be quite unreasonable to incur the extra expense of a safer alternative, otherwise that alternative should be used. This is the ALARP principle. This involves the assessor having a knowledge of the process alternatives at an early stage. In the light of this knowledge a reasoned judgement can be made about the likelihood of the process options being in compliance with the SAPs and ALARP in particular.

33. The objective is an ongoing technical dialogue between the regulator and the licensee to ensure that avoidable differences can be overcome at the conceptual stage. Because of the way in which the SAPs are applied in each set of circumstances relate to the plant under consideration, it is important that the underlying thinking of each side in the assessment process is clear. The investment by both sides should not be excessive and, in many cases, might involve a nominal time for a regular exchange on a routine basis.

34. To summarise the advantages and disadvantages of this approach:

Advantages:

a. there is an ongoing updating of thinking on the technical front as it develops on both sides
b. there is greater confidence that the licensee is pursuing a course of action which is not likely to require major changes at a later stage
c. there is enhanced confidence developed in the regulator that the licensee is giving safety an appropriate profile in the technical thinking in the light of modern commercial pressure

Disadvantages:

a. there will be an ongoing commitment from both sides on a regular basis
b. there will be a communication commitment both within the licensee's organisation and that of the regulator to ensure that those who need to know the early information get it quickly and accurately

Without current, formal, practical experience in these technical aspects it is not possible to balance the costs and benefits with any accuracy although such a balance becomes particularly important in the current financial climate in the UK. This becomes even more difficult when the outcome may be very many years away and standards can change in the gestation period. However, there have been a number of examples of successful liaison between NII and its licensees over many years for other purposes.

35. The value of close and early liaison between NII and its licensees has been highlighted at public inquiries. In the last decade or so these have included:

a. the Sizewell B public inquiry under Sir Frank Layfield. This followed many years of NII assessment for the PWR in the UK and Sir Frank stated in his report [8]:

"Confidence can be placed in the technical competence of the NII and CEGB in carrying out their respective responsibilities for the safety of the Sizewell B project."

b. the Hinkley Point C public inquiry under Michael Barnes QC who in his report said [9]:

"It is scarcely possible to exaggerate the importance of the judgement of the NII on matters of safety. ...... Their history, experience and organisation all combine to add weight to their judgement."

Such experience and judgement is only gained by continued critical testing of safety cases presented on the technical aspects of the plants proposed by licensees and an awareness of international developments.

c. the inquiry into UKAEA's proposal to build a European Demonstration Reprocessing plant at Dounreay, Thurso in Scotland held under Mr Bell. He reported [11]:

"It cannot therefore be a valid criticism of regulatory agencies that they discuss the safety implications of a project at an early stage; nor should exception be taken to such dialogue continuing under one statutory code while a lengthy PLI (public local inquiry) proceedings under a parallel planning code. The early involvement of NII in considerations of design safety is encouraged by government policy."

d. the recent review of potential candidate reactor designs undertaken in anticipation of the government's current review of the nuclear power industry. In this the NII undertook a review of the possible reactor designs which might be considered by the current (and possibly future) licensees. This extensive tranche of pre-emptive technical work should put both the industry and the regulator in a good position to speak authoritatively to the government's review.
in the current political climate it has become even more necessary to ensure that there
as few hindrances to the assessment process as is reasonable. In the recent past, NII
senior managers have met with their counterparts in the industry and there is a ongoing
commitment to ensure no undue delays in any safety submission. Obviously, if the
technical basis for any such submission is well known to the regulator and the licensee
is aware of the current regulatory thinking such difficulties can be minimised without
compromising safety or the integrity of either organisation.

Thus, in the widest sense, good liaison pays dividends for both the licensee and the regulator.
Some of these can be readily identified:

a. that each side is aware of developments and thereby avoidable difficulties can be
   resolved as soon as practical.

b. the licensee (or potential licensee) can be confident that investments can be undertaken
   with minimum financial risk

c. both organisations present a sound and competent face in public

Therefore, there is every reason to continue this work. In particular, for reprocessing plant the
chemical engineer plays a key technical role and should be involved in technical liaison. Also,
such a technical liaison should provide adequate information in support of the overall process
to engender confidence in the final conceptual design choices

36. The current generation of plants either in design and construction or in commissioning at
Sellafield are largely waste management plants with the significant exception of THORP.
Since BNFL's Sellafield site evolved from plant which had been initially intended for weapons
development there are a number of older plants which have reached the end of their useful
working lives. These are to be decommissioned on a rolling programme. However, records of
construction and operation which would have assisted in decommissioning are often sketchy
and some are missing. Therefore there can be significant uncertainties in decommissioning
justification which must be dealt with prudently. Assessment experience shows that there are
additional aspects which must be taken into account. The main ones include:

a. how the process conforms with NII policy. Most relevant is the policy that waste
   forms should be immobilised as soon as reasonably practical after their creation. This
   can be seen in action in the Windscale Vitrification Plant (WVP) and the recent cement
   encapsulation plants - all at Sellafield.

b. how the process fits into the overall UK national waste management strategy. Overall
   responsibility for waste strategy in the UK lies with the Department of the
   Environment and the Ministry of Agriculture, Fisheries and Food, known as the
   Authorising Departments (as they grant an Authorisation for waste disposal under their
   separate legislation). In decommissioning there must be a balance between the nuclear
   safety requirements and those for disposal.

c. overall integration into the site framework. Where there is considerable uncertainty
   about a waste feedstock and analysis is difficult (if not impossible), then any such plant
   must be designed to a very conservative specification which takes account of all
   reasonably foreseeable eventualities. The design concept is vital in such instances.
Unit B2 in NII deals with both the conceptual design stages for chemical plant and with waste management. This organisational feature has eased any potential assessment difficulty where there may be a clash of interest at the concept stage.

CONCLUSIONS

37. In Part I of this paper I have outlined the regulatory regime in the UK highlighting the ALARP principle, I have shown how from the thinking in Tolerability of Risk (TOR) there is a direct link into the Safety Assessment Principles for nuclear plant (SAPs) and how the HSE's Nuclear Safety Division, of which NII is a part, is well organised to do its work. I highlight the differences between legislation, approved codes of practice (ACOPs) and guidance. In Part II I have shown, in part, the process of assessment, how it may be applied to the design concept stage of a project and how an interchange on the licensee's current conceptual thinking and progress meets the needs of the licensee, the regulator and fulfils government policy.

38. I believe that for a small increment in effort there are benefits in involving the regulator, and the chemical engineering assessor in particular, at the conceptual stage of projects since this will minimise commercial risks for the licensee and engender regulator confidence. There is a past record of success in liaison between licensee and regulator in the UK and this experience indicates a positive cost benefit relationship in the medium to long term.

ACKNOWLEDGEMENT

39. This paper has drawn freely on the work of a number of colleagues. I would like to thank many colleagues in the Inspectorate for their help in preparing this paper. The views expressed are those of the author and are not necessarily the views of the NII.
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Figure 2: Nuclear Safety Division

UNIT 1 HSE/MSC, OGD LINKS, PARLIAMENTARY AND MINISTERIAL
   General Policy and Planning

UNIT 2 International Policy

UNIT 3 Information Technology

POLICY STRATEGY AND INTERNATIONAL

UNIT 1 Probabilistic Safety Analysis, Risk Assessment, Fault Analysis
   Human Factors

UNIT 3 Radioactive Waste, Chemistry, Chemical Engineering
   Documentation

BRANCH B

UNIT 3 Radiation Protection, Health Physics, Dosimetry Service
   Reactor Safety

ASSESSMENT (PHYSICS)

UNIT 4 Control & Instrumentation

UNIT 2 Transient Analysis, Pool Behaviour, Reactor Physics, Severe Accident Management

UNIT 6 PWR Projects, UKAEA Projects, BVR's Projects

BRANCH C

UNIT 1 Structural Integrity 1 PWR & BNF Plants, Steam pressure Vessels and Pipework

UNIT 2 Structural Integrity 1: Magnox AGR & UKAEA Plants Steam Pressure Vessels

UNIT 3 Civil Structures and Supporting Infrastructure
   Concrete Pressure Vessels, External Heaters - Natural and ex. made

ASSESSMENT (ENGINEERING)

UNIT 4 Licences Quality Assurance arrangements
   Quality Assurance Audits, Safety Management

UNIT 3 Mechanised and Electrical System Reliability of Mechanical and Electrical Plant
   Internal Heaters other than Structural Integrity of Steam Pressure Parts

UNIT 4 Magnox Long Term Safety Review, AGR Pool Dry boiler store
   Generic Problems Associated with Ageing and Plant Life Extension
   AGR, PFR & BETH/PLUTO Safety Cases

BRANCH D

UNIT 1 Inspections BNR Solicitor, Brig

UNIT 2 UKAEA Bicester, Harwell, Winfrith
   Windscale Sellafield, Springfield Sellafield
   Molt Venosa, Bessacarr

UNIT 2 Project: Project Inspection of new projects and major plant modifications on Bracco 3 Site

UNIT 4 Bosport, Borth, Balfour Beatty (Shib, Wylfa Borrow
   Al Amberata, Cardin, Harwell
   BNF Copesand, Springfield, Sellafield
   Research Reactor, Balfours, Med Grimsby

UNIT 1 Inspection Strategy, Advice & Response to OPA's
   Research Challenges and Generic Inspection Issues
   Emergency Planning

BRANCH E

UNIT 2 Berkeley, Stanford A & B
   MIDBAY P A & B Bradwell

UNIT 3 Harwell, Heysham 1 & 2, Torus
   Shawwell A, Shawwell B

UNIT 4 Calder Hall, Chapelcross
   Dungeness A & B Bradwell

BRANCH F

UNIT 1 Radiation Protection Policy, Radioactive Waste,
   Decommissioning, DOE liaison

UNIT 2 Emergency Arrangements,
   Fuels Reactor Systems, Incident Reporting, Siteing,
   Cost of Safety & CDA, Licensing, Management of Safety

UNIT 3 Nuclear Safety Research Management

205
APPENDIX I

Nuclear Site Licence

Standard Conditions
1. **INTERPRETATION**

(1) In the conditions set out in this Schedule to this licence, unless the context otherwise requires, the following expressions have the meanings hereby respectively assigned to them, that is to say -

"commissioning" means the process during which plant components and systems, having been constructed or modified, are made operational and verified to be in accordance with design assumptions and to have met the appropriate safety criteria;

"excepted matter" has the meaning assigned thereto in the Nuclear Installations Act 1965 (as amended) and the Nuclear Installations (Excepted Matter) Regulations 1978 made thereunder;

"the Executive" means the Health & Safety Executive;

"experiment" means any test or non-routine activity other than an activity carried out pursuant to conditions 21 and 28;

"installation" means "nuclear installation" and has the meaning assigned thereto in the Nuclear Installations Act 1965 (as amended);

"the licensee" and "the site" each has the meaning assigned thereto in paragraph 1 of this licence;

"modification" means any alteration to buildings, plants, operations, processes or safety cases and includes any replacement, refurbishment or repairs to existing buildings, plants or processes and alterations to the design of plants during the period of construction;

"nuclear matter" and "relevant site" each has the meaning assigned thereto in the Nuclear Installations Act 1965 (as amended);

"nuclear safety committee" means any nuclear safety committee established pursuant to condition 13 of this Schedule;

"operations" includes maintenance, examination, testing and operation of the plant and the treatment, processing, keeping, storing, accumulating or carriage of any radioactive material or radioactive waste and "operating" and "operational" shall be construed accordingly;

"radioactive material" and "radioactive waste" each has the meaning assigned thereto in the Radioactive Substances Act 1960;

"safety" refers to the safety of persons whether on or off the site;

"safety case" means the document or documents produced by the licensee in accordance with condition 14 of this Schedule.
(2) In these conditions except where the context otherwise requires -

(a) any reference to the singular shall include the plural and vice versa and any reference to the masculine shall include the feminine;

(b) any reference to any arrangement, agreement, approval, consent, direction, specification, notification or any formal communication between the Executive and the licensee (and vice versa) shall be deemed to be a reference to a written document;

(c) any reference to a numbered condition is a reference to the condition so numbered in this Schedule.

(3) Where in these conditions the Executive requires any matter to be approved or to be carried out only with its consent or to be carried out as it directs the Executive may -

(a) from time to time modify, revise or withdraw either wholly or in part any such approval, direction or consent;

(b) approve either wholly or in part any modification or revision or any proposed modification or revision to any matter for the time being approved.

2. MARKING OF THE SITE BOUNDARY

(1) The licensee shall make and implement adequate arrangements to prevent unauthorised persons from entering the site or, if so directed by the Executive, from entering such part or parts thereof as the Executive may specify.

(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 mark the boundaries of the site by fences or other appropriate means and any such fences or other means used for this purpose shall be properly maintained.

(5) The licensee shall, if so directed by the Executive, erect appropriate fences on the site in such positions as the Executive may specify and shall ensure that all such fences are properly maintained.

3. RESTRICTION ON DEALING WITH THE SITE

The licensee shall not convey, assign, transfer, let or part with possession of the site or any part thereof or grant any licence in relation thereto without the consent of the Executive.

4. RESTRICTIONS ON NUCLEAR MATTER ON THE SITE

(1) The licensee shall ensure that no nuclear matter is brought onto the site except in accordance with adequate arrangements made by the licensee for this purpose.
(2) The licensee shall ensure that no nuclear matter is stored on the site except in accordance with adequate arrangements made by the licensee for this purpose.

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

(4) 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.

(5) For new installations, if the Executive so specifies, the licensee shall ensure that no nuclear matter intended for use in connection with the new installation is brought onto the site for the first time without the consent of the Executive.

5. CONSIGNMENT OF NUCLEAR MATTER

(1) The licensee shall not consign nuclear matter (other than excepted matter and radioactive waste) to any place in the United Kingdom other than a relevant site except with the consent of the Executive.

(2) The licensee shall keep a record of all nuclear matter (including excepted matter and radioactive waste) consigned from the site and such record shall contain particulars of the amount, type and form of such nuclear matter, the manner in which it was packed, the name and address of the person to whom it was consigned and the date when it left the site.

(3) The licensee shall ensure that the aforesaid record is preserved for 30 years from the date of dispatch or such other period as the Executive may approve except in the case of any consignment or part thereof subsequently stolen, lost, jettisoned or abandoned, in which case the record shall be preserved for a period of 50 years from the date of such theft, loss, jettisoning or abandoning.

6. DOCUMENTS, RECORDS, AUTHORITIES AND CERTIFICATES

(1) The licensee shall make adequate records to demonstrate compliance with any of the conditions attached to this licence.

(2) Without prejudice to any other requirements of the conditions attached to this licence the licensee shall make and implement adequate arrangements to ensure that every document required, every record made, every authority, consent or approval granted and every direction or certificate issued in pursuance of the conditions attached to this licence is preserved for 30 years or such other periods as the Executive may approve.

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

(4) 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.
(5) The licensee shall furnish to the Executive copies of any such document, record, authority or certificate as the Executive may specify.

7. INCIDENTS ON THE SITE

(1) The licensee shall make and implement adequate arrangements for the notification, recording, investigation and reporting of such incidents occurring on the site:

(a) as is required by any other condition attached to this licence;

(b) as the Executive may specify; and

(c) as the licensee considers necessary.

(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.

8. WARNING NOTICES

The licensee shall ensure that suitable and sufficient notices are kept on the site for the purposes of informing persons thereon of each of the following matters, that is to say:

(a) the meaning of any warning signal used on the site;

(b) the location of any exit from any place on the site, being an exit provided for use in the event of an emergency;

(c) the measures to be taken by such persons in the event of fire breaking out on the site or in the event of any other emergency;

and that such notices are kept posted in such positions and in such characters as to be conveniently read by those persons.

9. INSTRUCTIONS TO PERSONS ON THE SITE

The licensee shall ensure that every person authorised to be on the site receives adequate instructions (to the extent that this is necessary having regard to the circumstances of that person being on the site) as regards the risks and hazards associated with the plant and its operation, the precautions to be observed in connection therewith and the action to be taken in the event of an accident or emergency on the site.

10. TRAINING

(1) The licensee shall make and implement adequate arrangements for suitable training of all those on site who have responsibility for any operations which may affect safety.
(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.

11. EMERGENCY ARRANGEMENTS

(1) Without prejudice to any other requirements of the conditions attached to this licence the licensee shall make and implement adequate arrangements for dealing with any accident or emergency arising on the site and their effects.

(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) Where any such arrangements require the assistance or co-operation of, or render it necessary or expedient to make use of the services of any person, local authority or other body the licensee shall ensure that each person, local authority or other body is consulted in the making of such arrangements.

(5) The licensee shall ensure that such arrangements are rehearsed at such intervals and at such times and to such extent as the Executive may specify or, where the Executive has not so specified, as the licensee considers necessary.

(6) The licensee shall ensure that such arrangements include procedures to ensure that all persons in his employ who have duties in connection with such arrangements are properly instructed in the performance of the same, in the use of the equipment required and the precautions to be observed in connection therewith.

12. Duly authorised and other suitably qualified and experienced persons

(1) The licensee shall make and implement adequate arrangements to ensure that only suitably qualified and experienced persons perform any duties which may affect the safety of operations on the site or any duties assigned by or under these conditions or any arrangements required under these conditions.

(2) The aforesaid arrangements shall also provide for the appointment, in appropriate cases, of duly authorised persons to control and supervise operations which may affect plant safety.

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

(4) 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.
(5) The licensee shall ensure that no person continues to act as a duly authorised person if, in the opinion of the Executive, he is unfit to act in that capacity and the Executive has notified the licensee to that effect.

13. NUCLEAR SAFETY COMMITTEE

(1) The licensee shall establish a nuclear safety committee or committees to which it shall refer for consideration and advice the following:

(a) all matters required by or under these conditions to be referred to a nuclear safety committee;

(b) such arrangements or documents required by these conditions as the Executive may specify and any subsequent alteration or amendment to such specified arrangements or documents;

(c) any matter on the site affecting safety on or off the site which the Executive may specify; and

(d) any other matter which the licensee considers should be referred to a nuclear safety committee.

(2) The licensee shall submit to the Executive for approval the terms of reference of any such nuclear safety committee and shall not form a nuclear safety committee without the aforesaid approval.

(3) The licensee shall ensure that once approved no alteration or amendment is made to the terms of reference of such a nuclear safety committee unless the Executive has approved such alteration or amendment.

(4) The licensee shall appoint at least seven persons as members of a nuclear safety committee including one or more members who are independent of the licensee's operations and shall ensure that at least five members are present at each meeting including at least one independent member.

(5) The licensee shall furnish to the Executive the name, qualifications, particulars of current posts held and the previous relevant experience of every person whom he appoints as a member of any nuclear safety committee forthwith after making such appointment. Notwithstanding such appointment the licensee shall ensure that a person so appointed does not remain a member of any nuclear safety committee if the Executive notifies the licensee that it does not agree to the appointment.

(6) The licensee shall ensure that the qualifications, current posts held and previous relevant experience of the members of any such committee, taken as a whole, are such as to enable that committee to consider any matter likely to be referred to it and to advise the licensee authoritatively and, so far as practicable, independently.
(7) The licensee shall ensure that a nuclear safety committee shall consider or advise only during the course of a properly constituted meeting of that committee.

(8) The licensee shall send to the Executive within 14 days of any meeting of any such committee a full and accurate record of all matters discussed at that meeting including in particular any advice given to the licensee.

(9) The licensee shall furnish to the Executive copies of any document or any category of documents considered at any such meetings that the Executive may specify.

(10) The licensee shall notify the Executive as soon as practicable if it is intended to reject, in whole or in part, any advice given by any such committee together with the reasons for such rejection.

(11) Notwithstanding paragraph (7) of this condition, where it becomes necessary to obtain consideration of, or advice on, urgent safety proposals (which would normally be considered by a nuclear safety committee) the licensee may do so in accordance with appropriate arrangements made for the purpose by the licensee, considered by the relevant nuclear safety committee and approved by the Executive.

(12) The licensee shall ensure that once approved no alteration or amendment is made to the approved arrangements described in paragraph (11) of this condition unless the relevant nuclear safety committee has considered and the Executive has approved such alteration or amendment.

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.

15. PERIODIC REVIEW

(1) The licensee shall make and implement adequate arrangements for the periodic and systematic review and reassessment of safety cases.

(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, if so directed by the Executive, carry out a review and reassessment of safety and submit a report of such review and reassessment to the Executive at such intervals, within such a period and for such of the matters or operations as may be specified in the direction.

16. SITE PLANS, DESIGNS AND SPECIFICATIONS

(1) The licensee shall submit to the Executive an adequate plan of the site (hereinafter referred to as the site plan) showing the location of the boundary of the licensed site and every building or plant on the site which might affect safety.

(2) The licensee shall submit to the Executive with the site plan a schedule giving particulars of each such building and plant thereon and the operations associated therewith.

(3) If any changes are made on the site which affect the said buildings, plant or operations, the licensee shall forthwith send an amended site plan and schedule to the Executive incorporating these changes.

(4) The licensee shall furnish to the Executive such plans, designs, specifications or other information relating to such buildings, plants and operations as the Executive may specify.

17. QUALITY ASSURANCE

(1) Without prejudice to any other requirements of the conditions attached to this licence the licensee shall make and implement adequate quality assurance arrangements in respect of all matters which may affect safety.

(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 such copies of records or documents made in connection with the aforesaid arrangements as the Executive may specify.

18. RADIOLOGICAL PROTECTION

(1) The licensee shall make and implement adequate arrangements for the assessment of the average effective dose equivalent (including any committed effective dose equivalent) to such class or classes of persons as may be specified in the aforesaid arrangements and the licensee shall forthwith notify the Executive if the average effective dose equivalent to such class or classes of persons exceeds such level as the Executive may specify.
(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.

19. CONSTRUCTION OR INSTALLATION OF NEW PLANT

(1) Where the licensee proposes to construct or install any new plant which may affect safety the licensee shall make and implement adequate arrangements to control the construction or 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 aforesaid arrangements shall where appropriate divide the construction or installation into stages. Where the Executive so specifies the licensee shall not commence nor thereafter proceed from one stage to the next of the construction or installation without the consent of the Executive. The arrangements shall include a requirement for the provision of adequate documentation to justify the safety of the proposed construction or installation and shall where appropriate provide for the submission of this documentation to the Executive.

(5) The licensee shall, if so directed by the Executive, halt the construction or installation of a plant and the licensee shall not recommence such construction or installation without the consent of the Executive.

20. MODIFICATION TO DESIGN OF PLANT UNDER CONSTRUCTION

(1) The licensee shall ensure that no modification to the design which may affect safety is made to any plant during the period of construction except in accordance with adequate arrangements made and implemented by the licensee for that purpose.

(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 aforesaid arrangements shall provide for the classification of modifications according to their safety significance. The arrangements shall where appropriate divide modifications into stages. Where the Executive so specifies the licensee shall not commence nor thereafter proceed from one stage to the next of the modification without the consent of the Executive. The arrangements shall include a requirement for the provision of adequate documentation to justify the safety of the proposed modification and shall where appropriate provide for the submission of this documentation to the Executive.
21. **COMMISSIONING**

(1) The licensee shall make and implement adequate arrangements for the commissioning of any plant or process which may affect safety.

(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 aforesaid arrangements shall where appropriate divide the commissioning into stages. Where the Executive so specifies the licensee shall not commence nor thereafter proceed from one stage to the next of the commissioning without the consent of the Executive. The arrangements shall include a requirement for the provision of adequate documentation to justify the safety of the proposed commissioning and shall where appropriate provide for the submission of this documentation to the Executive.

(5) The licensee shall appoint a suitably qualified person or persons for the purpose of controlling, witnessing, recording and assessing the results of any tests carried out in accordance with the requirements of the aforesaid commissioning arrangements.

(6) The licensee shall ensure that full and accurate records are kept of the results of every test and operation carried out in pursuance of this condition.

(7) The licensee shall ensure that no plant or process which may affect safety is operated (except for the purpose of commissioning) until:

   (a) the appropriate stage of commissioning has been completed and a report of such commissioning, including any results and assessments of any tests as may have been required under the commissioning arrangements referred to in paragraph (1) of this condition, has been considered in accordance with those arrangements; and

   (b) a safety case or cases as appropriate, which shall include the safety implications of modifications made since the commencement of construction of the plant and those arising from the commissioning of the plant, and any matters whereby the operation of the plant may be affected by such modifications or commissioning, has been considered in accordance with the arrangements referred to in paragraph (1) of this condition.

(8) The licensee shall, if so notified by the Executive, submit to the Executive the safety case for the aforesaid plant or processes prepared in pursuance of paragraph (7) of this condition and shall not commence operation of the relevant plant or process without the consent of the Executive.

22. **MODIFICATION OR EXPERIMENT ON EXISTING PLANT**

(1) The licensee shall make and implement adequate arrangements to control any modification or experiment carried out on any part of the existing plant or processes which may affect safety.
(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 aforesaid arrangements shall provide for the classification of modifications or experiments according to their safety significance. The arrangements shall where appropriate divide the modification or experiment into stages. Where the Executive so specifies the licensee shall not commence nor thereafter proceed from one stage to the next of the modification or experiment without the consent of the Executive. The arrangements shall include a requirement for the provision of adequate documentation to justify the safety of the proposed modification or experiment and shall where appropriate provide for the submission of the documentation to the Executive.

5. The licensee shall, if so directed by the Executive, halt the modification or experiment and the licensee shall not recommence such modification or experiment without the consent of the Executive.

23. OPERATING RULES

(1) The licensee shall, in respect of any operation that may affect safety, produce an adequate safety case to demonstrate the safety of that operation and to identify the conditions and limits necessary in the interests of safety. Such conditions and limits shall hereinafter be referred to as operating rules.

(2) The licensee, where the Executive so specifies, shall refer the operating rules arising from paragraph (1) of this condition to the relevant nuclear safety committee for consideration.

(3) The licensee shall ensure that operations are at all times controlled and carried out in compliance with such operating rules. Where the person appointed by the licensee for the purposes of condition 26 identifies any matter indicating that the safety of any operation or the safe condition of any plant may be affected that person shall bring that matter to the attention of the licensee forthwith who shall take appropriate action and ensure the matter is then notified, recorded, investigated and reported in accordance with arrangements made under condition 7.

(4) The licensee shall submit to the Executive for approval such of the aforesaid operating rules as the Executive may specify.

(5) The licensee shall ensure that once approved no alteration or amendment is made to any approved operating rule unless the Executive has approved such alteration or amendment.

(6) Notwithstanding the preceding provisions of this condition the Executive may, if in its opinion circumstances render it necessary at any time, agree to the temporary suspension of any approved operating rule.

217
24. OPERATING INSTRUCTIONS

(1) The licensee shall ensure that all operations which may affect safety are carried out in accordance with written instructions hereinafter referred to as operating instructions.

(2) The licensee shall ensure that such operating instructions include any instructions necessary in the interests of safety and any instructions necessary to ensure that any operating rules are implemented.

(3) The licensee shall, if so specified by the Executive, furnish to the Executive copies of such operating instructions and when any alteration is made to the operating instructions furnished to the Executive, the licensee shall ensure that such alteration is furnished to the Executive within such time as may be specified.

(4) The licensee shall make and implement adequate arrangements for the preparation, review and amendment of such operating instructions.

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

(6) 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.

25. OPERATIONAL RECORDS

(1) The licensee shall ensure that adequate records are made of the operation, inspection and maintenance of any plant which may affect safety.

(2) The aforesaid records shall include records of the amount and location of all radioactive material, including nuclear fuel and radioactive waste, used, processed, stored or accumulated upon the site at any time.

(3) The licensee shall record such additional particulars as the Executive may specify.

(4) The licensee shall furnish to the Executive such copies of extracts from such records at such times as the Executive may specify.

26. CONTROL AND SUPERVISION OF OPERATIONS

The licensee shall ensure that no operations are carried out which may affect safety except under the control and supervision of suitably qualified and experienced persons appointed for that purpose by the licensee.

27. SAFETY MECHANISMS, DEVICES AND CIRCUITS

The licensee shall ensure that a plant is not operated, inspected, maintained or tested unless suitable and sufficient safety mechanisms, devices and circuits are properly connected and in good working order.
EXAMINATION, INSPECTION, MAINTENANCE AND TESTING

(1) The licensee shall make and implement adequate arrangements for the regular and systematic examination, inspection, maintenance and testing of all plant which may affect safety.

(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 aforesaid arrangements shall provide for the preparation of a plant maintenance schedule for each plant. The licensee shall submit to the Executive for its approval such part or parts of any plant maintenance schedule as the Executive may specify.

(5) The licensee shall ensure that once approved no alteration or amendment is made to any approved part of any plant maintenance schedule unless the Executive has approved such alteration or amendment.

(6) The licensee shall ensure in the interests of safety that every examination, inspection, maintenance and test of a plant or any part thereof is carried out:

(a) by suitably qualified and experienced persons;

(b) in accordance with schemes laid down in writing;

(c) within the intervals specified in the plant maintenance schedule; and

(d) under the control and supervision of a suitably qualified and experienced person appointed by the licensee for that purpose.

(7) Notwithstanding the above paragraphs of this condition the Executive may agree to an extension of any interval specified in the plant maintenance schedule.

(8) When any examination, inspection, maintenance or test of any part of a plant reveals any matter indicating that the safe operation or safe condition of that plant may be affected, the suitably qualified and experienced person appointed to control or supervise any such examination, inspection, maintenance or test shall bring it to the attention of the licensee forthwith who shall take appropriate action and ensure the matter is then notified, recorded, investigated and reported in accordance with arrangements made under condition 7.

(9) The licensee shall ensure that a full and accurate report of every examination, inspection, maintenance or test of any part of a plant indicating the date thereof and signed by the suitably qualified and experienced person appointed by the licensee to control and supervise such examination, inspection, maintenance or test is made to the licensee forthwith upon completion of the said examination, inspection, maintenance or test.
29. DUTY TO CARRY OUT TESTS, INSPECTIONS AND EXAMINATIONS

(1) The licensee shall carry out such tests, inspections and examinations in connection with any plant (in addition to any carried out under condition 28 above) as the Executive may, after consultation with the licensee, specify.

(2) The licensee shall furnish the results of any such tests, inspections and examinations carried out in accordance with paragraph (1) of this condition to the Executive as soon as practicable.

30. PERIODIC SHUTDOWN

(1) When necessary for the purpose of enabling any examination, inspection, maintenance or testing of any plant or process to take place, the licensee shall ensure that any such plant or process shall be shut down in accordance with the requirements of its plant maintenance schedule referred to in condition 28.

(2) Notwithstanding paragraph (1) of this condition the Executive may agree to an extension of a plant's operating period.

(3) The licensee shall, if so specified by the Executive, ensure that when a plant or process is shut down in pursuance of paragraph (1) of this condition it shall not be started up again thereafter without the consent of the Executive.

31. SHUTDOWN OF SPECIFIED OPERATIONS

(1) The licensee shall, if so directed by the Executive, shut down any plant, operation or process on the site within such period as the Executive may specify.

(2) The licensee shall ensure that when a plant, operation or process is shut down in pursuance of paragraph (1) of this condition it shall not be started up again thereafter without the consent of the Executive.

32. ACCUMULATION OF RADIOACTIVE WASTE

(1) The licensee shall make and implement adequate arrangements for minimising so far as is reasonably practicable the rate of production and total quantity of radioactive waste accumulated on the site at any time and for recording the waste so accumulated.

(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) Without prejudice to paragraph (1) of this condition the licensee shall ensure that radioactive waste accumulated or stored on the site complies with such limitations as to quantity, type and form as may be specified by the Executive.
(5) The licensee shall, if so specified by the Executive, not accumulate radioactive waste except in a place and in a manner approved by the Executive.

33. DISPOSAL OF RADIOACTIVE WASTE

The licensee shall, if so directed by the Executive, ensure that radioactive waste accumulated or stored on the site is disposed of as the Executive may specify and in accordance with a authorisation granted under the Radioactive Substances Act 1960 or, as the case may be, the Radioactive Substances Act 1993.

34. LEAKAGE AND ESCAPE OF RADIOACTIVE MATERIAL AND RADIOACTIVE WASTE

(1) The licensee shall ensure, so far as is reasonably practicable, that radioactive material and radioactive waste on the site is at all times adequately controlled or contained so that it cannot leak or otherwise escape from such control or containment.

(2) Notwithstanding paragraph (1) of this condition the licensee shall ensure, so far as is reasonably practicable, that no such leak or escape of radioactive material or radioactive waste can occur without being detected, and that any such leak or escape is then notified, recorded, investigated and reported in accordance with arrangements made under condition 7.

(3) Nothing in this condition shall apply to discharges or releases of radioactive waste in accordance with an approved operating rule or with disposal authorisations granted under the Radioactive Substances Act 1960 or as the case may be the Radioactive Substances Act 1993.

35. DECOMMISSIONING

(1) The licensee shall make and implement adequate arrangements for the decommissioning of any plant or process which may affect safety.

(2) The licensee shall make arrangements for the production and implementation of decommissioning programmes for each plant.

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

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

(5) The aforesaid arrangements shall where appropriate divide the decommissioning into stages. Where the Executive so specifies the licensee shall not commence nor thereafter proceed from one stage to the next of the decommissioning without the consent of the Executive. The arrangements shall include a requirement for the provision of adequate documentation to justify the safety of the proposed decommissioning and shall where appropriate provide for the submission of this documentation to the Executive.
(6) The licensee shall, if so directed by the Executive where it appears to them to be in the interests of safety, commence decommissioning in accordance with the aforesaid arrangements and decommissioning programmes.

(7) The licensee shall, if so directed by the Executive, halt the decommissioning of a plant and the licensee shall not recommence such decommissioning without the consent of the Executive.
THE REGULATION BY HMNIII OF MAGNOX WASTE

RETRIEVAL PROJECTS AT SELLAFIELD

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OECD / NEA
Cadarache
France

20 & 21 September, 1994

223
SUMMARY

This paper examines the regulation by the Health and Safety Executive's Nuclear Installations Inspectorate of waste retrieval projects at Sellafield.

Drawing on two examples of current projects, I analyse some safety aspects related to waste retrieval and identify generic regulatory lessons learnt. Of central importance is the need to ensure that waste forms are chemically and physically stable, and so suitable for safe long term storage.

HM Nuclear Installations Inspectorate has gained valuable experience through its regulation of these projects. I set out the key regulatory lessons learnt in the hope that this experience will benefit members of the OECD/NEA Working Group on Fuel Cycle Safety.

Note

The views in this paper are those of the author and do not necessarily represent those of HM Nuclear Installations Inspectorate.
INTRODUCTION

This paper explains the work carried out by Her Majesty's Nuclear Installations Inspectorate (NII), part of the Health and Safety Executive (HSE), to regulate projects associated with the post operational clean out and decommissioning of intermediate level Magnox waste storage facilities at Sellafield, operated by British Nuclear Fuels plc (BNFL). Several projects are currently in progress, or are planned, to retrieve this waste. Through examples of these projects, the paper discusses the safety significance of the waste retrieval work and identifies the key regulatory lessons learnt.

The basis of UK health and safety law is that the persons who create a risk to people's health and safety are responsible for ensuring that the risk is properly controlled. The role of NII, on behalf of HSE, is to ensure that the operators of nuclear installations discharge this duty properly. The main legislation governing the safety of nuclear installations is the Health and Safety at Work etc Act, 1974, and the relevant statutory provisions of the Nuclear Installations Act, 1965, as amended. NII's regulatory powers derive from these Acts of Parliament. NII regulates the safety of nuclear installations via a Site Licence under the Nuclear Installations Act. My colleague Dr Trimble has prepared a complementary Working Group paper (1) which deals more fully with the legal basis of NII's work, and I invite the reader to consult this paper for further information on this topic.

NII has been concerned that certain high-hazard waste storage arrangements are being prolonged for longer than previously envisaged. Over a period of time NII has applied pressure to BNFL to develop a strategy for safe long term storage. BNFL in turn came to realise that they could not sustain a long term safety justification for current storage arrangements, and indeed that there were business-related incentives to improve levels of safety at the plants in question. As a result, an acceptable strategy is now established:

- retrieval of the waste;
- immobilisation in a cement matrix;
- recoverable storage in engineered facilities;
- long term storage in a suitable repository.
Immobilisation and storage are not discussed further in this paper, which concentrates on the first step of waste retrieval. But first, I summarise NII's approach to the regulation of projects.

REGULATION OF PROJECTS ON NUCLEAR CHEMICAL PLANT

When an operator of a nuclear installation wishes to carry out any activity on its site which might affect nuclear safety, it should assess the safety significance of that activity in accordance with its arrangements under the Site Licence. The operator should then prepare a safety case, usually in a standard format, and for all those activities which have more than a minor bearing on nuclear safety the operator should send the safety case to NII. [NII also has the right to review all safety cases, and to challenge the operator's categorisation of safety significance.]

These activities are referred to as projects. There are essentially three types of projects:

- New plant;
- Modifications to existing plant;
- Post-operational clean out and decommissioning.

Within NII's organisation, each project is handled by an individual inspector, whose task is to regulate that project in accordance with the site licence, to arrange specialist assessment of the safety case where appropriate, and to inspect the project for compliance with the Site Licence and other relevant statutory provisions. Regulation in this context includes the preparation and issue of enforcement notices and actions under the licence (formal instruments which are the exercise of NII's legal powers and allow operators to carry out given tasks or forbid certain actions), the investigation of incidents, and so on. Another key task for the project inspector is to liaise with the Authorising Departments, who have responsibility for authorising discharges of radioactivity from nuclear installations.

Each project can be sub-divided into a number of stages, dependent upon its complexity. NII singles out installation and active commissioning of projects as the two key stages. Installation, because NII needs to be satisfied that the operator has an adequate safety case before any permanent work is started. And active commissioning, because this is the first time that any radioactivity is introduced on a project, and NII must be satisfied that safety documentation demonstrates that all safety systems have been properly identified and will be thoroughly challenged during active commissioning before this stage commences.

Both of the examples of projects which follow fall into two of the three categories listed above. They are properly waste retrieval projects, in that their objective is to retrieve intermediate level waste from existing storage facilities which do not represent safe
long term storage options, and also require major modifications to existing facilities. On both counts, each represents a significant challenge to safety in its own right.

MAGNOX WASTE RETRIEVAL PROJECTS AT SELLAFIELD

Background

Since the 1950s, the reprocessing of irradiated Magnox fuel at Sellafield has given rise to a considerable volume of intermediate level waste (ILW) in a mobile form. The ILW of special interest to this paper is the Magnox cladding, which is removed prior to reprocessing of the irradiated fuel bar. Until recently the stripped cladding was stored under water in silos. Magnox swarf corrodes under water in an exothermic reaction giving rise to an essentially insoluble residue. The heat generated, and the associated hydrogen evolved, have been significant safety concerns over the years, and instances are well documented.

The form that the ILW takes is a matter of particular regulatory concern. The material is physically and chemically unstable, and has the potential for giving rise to a significant radiological release should an incident cause a breach of containment. BNFL has set in train a major programme of post operational clean out and decommissioning work to significantly reduce the contribution to overall site risk represented by the plants. This is achieved by recovering and stabilising the waste. However, the retrieval process itself requires careful consideration to make sure that the act of retrieval does not in itself increase the risk associated with a plant.

Examples

Using examples of current waste retrieval projects, I describe below the measures which BNFL have taken to ensure that waste retrieval is carried out in a safe and controlled manner, and the basis for the regulatory lessons learnt which I identify later. I start each example with a brief description of the function of the relevant buildings. This will aid the understanding of the nature of the wastes and the way in which they have been generated. I describe the source of Magnox waste and the reasons why, over a period of years, the management of waste storage gave rise to the problems experienced today. I deal with two specific forms of Magnox waste, namely sludge and swarf - I define these terms in the text. Note that sludge and swarf are "Intermediate Level Waste" in terms of the UK classification of waste (see Annex 1). Two Figures at the end of this paper show the basic layout and arrangement of B30 and B38, where the examples are based, buildings which are located in an area of Sellafield known as Ponds West. The description of buildings and projects which follow are not intended to be detailed, but should be sufficient to gain an overall feel for the topics, and to explain the regulatory issues.
EXAMPLE 1 - B30, Magnox Ponds and Decanning Facility

Retrieval of sludge from D-Bay

Plant Description

B30 was commissioned in 1959/60 for the purpose of receiving and decanning Magnox fuel, and for exporting irradiated fuel rods for reprocessing in the Chemical Separation Plant B205. B30 operated until 1986, when its function was taken over by the Fuel Handling Plant, B311.

Spent Magnox fuel entered B30 via the Inlet Building and was placed in skips in the Main Pond for a period of cooling, to allow shorter lived isotopes to decay to acceptable levels. Originally, the fuel was then moved to the Wet Bay Building, to allow decanning of the fuel rods; subsequently decanning operations were carried out in the newer Decanning Building within B30. The fuel rods were then flanked to B205, and the remaining fuel cans (known as swarf) taken for storage under water at B38, the solid active waste storage facility.

The licensees opted for storage of Magnox clad fuel under water to minimise the likelihood of fire if Magnox and irradiated uranium were exposed to air. Unfortunately, Magnox cladding corrodes in water to form an essentially insoluble residue mostly made up of magnesium hydroxide, and commonly referred to as sludge, and this corrosion reaction evolves hydrogen. The amount of sludge in the Main Pond of B30 built up slowly, and gave rise to solids in suspension, and in turn to higher background radiation and reduced visibility in the ponds, which further slowed the rate of decanning because of extended storage times. This led to increased amounts of corrosion, and hence to more sludge generation - a vicious circle. By the time that the B311 took over B30's role, B30 had accumulated over one thousand cubic metres of sludge. B30 also contains several hundred tonnes of fuel awaiting reprocessing.

The current radiological conditions in B30 are poor, both in terms of general building radiation levels and levels of airborne activity. The potential exists for personnel to exceed statutory dose limits unless special preventive measures are applied. However BNFL have now embarked on a series of plant improvements and post operational clean out (POCO) projects. The objective of POCO is to remove all fuel, sludge and debris, as soon as reasonably practicable, and so reduce levels of radiation significantly, to enable further decommissioning to proceed safely. This objective is complicated by the fact that the layout of B30 is complex and access is physically restricted. NII has reviewed the overall programme of POCO activities and has sampled a number for specific regulation, of which the most significant is D-Bay Desludging - the retrieval of sludge from D-Bay.
Project Description

B30, Retrieval of Sludge from D-Bay

D-Bay is full of sludge, with a volume about 230 m³. It has also sustained a crack in its south wall, thought to be due to early thermal shrinkage shortly after the concrete wall was cast, and as a consequence, it is to be the first wet bay to be de-sludged. BNFL have designed the bay sludge retrieval facility (BSRF) to perform the de-sludging. The BSRF achieves transfer of sludge by supplying motive water to a fixed jet pump which induces sludge into a transfer pipeline. Five redistribution pumps move the sludge towards the jet pump. Sludge is pumped through a shielded co-axial transfer pipeline to settling tanks, and ultimately the sludge is encapsulated.

The equipment which makes up the BSRF is available "off the shelf", although not necessarily in the configuration required by BNFL. The real area of difficulty in this project lies in the ability to install and commission the facility safely in an area where the levels of radiation and contamination are high and where access is restricted. This calls for BNFL to carry out a thorough study of methods of work to ensure that dose uptake is "as low as reasonably practicable" - the ALARP principle, a prominent feature in UK health and safety law which requires a balance to be drawn between the benefits of a given task and the costs incurred in reducing still further the risks associated with the task. The ALARP studies require a knowledge of the working environment, and a precise breakdown of specific operations. The culmination of this work is a series of Work Control Documents, laying down the steps to be taken and identifying those responsible for given actions. In this way BNFL minimise uptake of radiation doses.

BNFL tested the BSRF extensively at UKAEA's Winfrith site and at a fifth-scale model rig housed at Sellafield. BNFL then assembled the equipment off-site, at the manufacturer's works, to conduct inactive commissioning tests. This latter facility enabled operating and maintenance personnel to gain a first hand impression of the BSRF in totally inactive conditions, whilst at the same time proving the equipment and demonstrating the correct operation of safety systems. The use of off-site facilities to eliminate dose uptake during inactive commissioning has been a prominent feature of the Sellafield waste retrieval projects, one which BNFL wholeheartedly supports.

BNFL has completed inactive commissioning of the BSRF successfully, and installation is underway. Active commissioning is due to start shortly.
EXAMPLE 2 - B38, Solid Active Waste Storage Facility

Retrieval of swarf from Compartments 19 - 22

Plant Description

Swarf has been stored underwater in concrete compartments at B38 from 1964 until recently, when the Magnox Encapsulation Plant B389 started receiving swarf directly from B311 for encapsulation. The original B38 consisted of six single-contained compartments (numbered 1 to 6). The first extension (compartments 7 to 12), which includes a double-skin construction and the facility for monitoring leakage, is separated from the original building by a void. A second and third extension (compartments 13 & 14, and 15 to 22 respectively) were added, and are double contained. Compartments 1 to 10, 12 to 14 and 16 to 22 contain swarf under water. The compartments also store various other forms of intermediate level waste. See Figure 2.

The risk of a fire due to the generation of hydrogen as a result of swarf corrosion is a dominant feature of the safety case for B38, because the compartments are fully contained. The chemical reaction is exothermic and, if unchecked, can cause a runaway-type excursion. Because of this, the first, second and third extensions of B38 have been fitted (or back-fitted) with cooling and inerting capabilities. This means that swarf in the later compartments is relatively intact, and so presents the greatest risk of an excursion. Little intact swarf remains in the original part of B38, and so cooling and inerting is not installed.

The other significant aspect of B38 is the potential for leakage of water from the compartments. A major leak occurred in 1976. NII carried out an investigation of this leak, and published their findings (Ref 2). The report required BNFL to develop methods for retrieval of waste and for reprocessing of swarf as soon as practicable after decanning in preference to continued storage under water. In other words, the incident and NII's report helped pave the way for swarf retrieval and encapsulation. The incident also helped clarify NII's stance with respect to interim storage: NII has resisted attempts to build further extensions to B38, and persuaded BNFL to adopt the strategy of immobilisation and the creation of engineered stores for encapsulated ILW.

I have already mentioned the importance which NII attaches to the retrieval of swarf from B38. The potential for hydrogen generation leading to an excursion renders B38 one of the more hazardous buildings at Sellafield. NII's refusal to allow BNFL to build further interim storage capability forced BNFL down the road to a safer, long term storage option, namely immobilisation by encapsulation. And the clean up of B38 by retrieval of the contents of the compartments became a priority. BNFL has drawn up a strategy to empty B38, and waste retrieval has started with the recovery of swarf from Compartment 19 in the third extension.
B38, Retrieval of Swarf from Compartments 19 to 22

It was agreed that BNFL would first retrieve the essentially intact swarf from the third (newest) extension, because this swarf represents the greatest risk of an excursion. No equipment existed at the time to fulfill this objective, and BNFL commissioned a novel plant named the swarf retrieval facility (SRF). In essence, the design principles of the SRF were that it should have the same "footprint" as those existing B38 machines which service the compartments, be capable of retrieving swarf and keep it under water for export using a standard flask, and to effect a seal so that inerting of a compartment can be carried out in the event of detection of hydrogen.

The basis of the SRF is a simple petal grab which is lowered into a compartment, collects a quantity of swarf, and then deposits the swarf into a water-filled bin. When full, the bin is raised into a flask which is in turn transported to the Magnox Encapsulation Plant. BNFL designed the SRF in a modular form to accommodate existing B38 craneage. The substantial weight of the SRF represents a significant additional load for the building to support and necessitated verification that the building could support the SRF. The seismic capability of B38 also had to be re-examined to make sure that the SRF did not affect it significantly. [Note that only the third extension of B38 has designed seismic capability; BNFL assessed the seismic capability of the rest of the building retrospectively.]

The SRF was fabricated at works, and subjected to an exhaustive inactive commissioning programme, carried out off-site. Once again, this provided an ideal opportunity for ironing out any problems without attracting any dose detriment, and for training operators and maintenance personnel.

BNFL has retrieved over 450 bins of swarf to date, equivalent to about one half of the contents of one compartment. The swarf retrieved to date is generally in very good condition, emphasising the role played by the cooling circuits on the third extension in arresting the Magnox corrosion reaction. NII has now issued to BNFL formal notification under the Site Licence that they can operate the SRF, marking the successful completion of active commissioning. Thereafter, BNFL will be able to complete the emptying of Compartment 19, and then transfer the SRF to the remaining third extension compartments 20 to 22. In the near future, BNFL will present to NII its plans for post operational clean out of the remaining eighteen compartments in B38.
REGULATORY LESSONS LEARNT

Several issues have emerged which are common to all waste retrieval projects, and their identification will aid the assessment of future post operational clean out and decommissioning activities. These are arranged below, in no particular order of importance:

i) Waste Retrieval and Immobilisation. Ideally, any hazardous waste should be rendered safe at source. However, Magnox sludge and swarf has existed for some years mainly in a form which presents a potential threat due to its physical and chemical characteristics. Clearly the top priority for dealing with such waste is to retrieve and immobilise the waste as soon as practicable, and so render it suitable for safe long term storage. BNFL has recognised the importance of this work, developed a strategy for cleaning up the plants in question, and has started retrieval of the waste. Progress in this strategic development was influenced significantly by NII's refusal to allow BNFL to build any further interim storage capability.

ii) The Role of the Regulator. As a result of NII's decision not to allow further silo storage, BNFL evolved a satisfactory strategy for dealing with Magnox waste. At this point, both regulator and regulated were committed to cleaning out the plants and encapsulating the waste, hence both began to share the same objective. In such situations, provided the plant operator maintains adequate progress and demonstrates adequate control of the safety of its work, I consider that the regulator should adopt a supportive role where appropriate.

iii) Regulatory Standards. For the two waste retrieval projects which I have described above, the buildings do not represent satisfactory long term storage arrangements, and the situation is getting worse. This calls for a balanced approach to regulation, wherein the regulator should apply appropriate standards based on ALARP, although modern standards remain as a benchmark from which to base these regulatory judgements. I apply this principle when regulating the Sellafield waste retrieval projects: my objective is to ensure that BNFL remove the source of the risk from the waste storage facilities as soon as practicable, whilst at the same time carrying out waste retrieval safely.

iv) Risks associated with modifications. Carrying out modifications to effect waste retrieval from existing plant with poor radiological conditions inevitably implies that the risk from that plant will increase temporarily as work is carried out, before achieving improved steady state conditions. Therefore the regulator should ensure that the licensee has demonstrated that there is an overall benefit to be gained by performing any particular modification.

v) Safety during Installation and Commissioning. The physical act of performing a modification on plant to enable waste retrieval to commence is often a
significant contributor to safety, sometimes more significant than a theoretical fault analysis. Very often the risk to the work-force is dominated by "conventional" safety concerns. This means that the operator should pay special attention to safety during installation and active commissioning. The regulator should ensure that the operator has carried out a systematic study of potential hazards and that they have in place effective managerial controls.

vi) Design Safety Case. None of the Ponds West plants in question were designed with retrieval of waste in mind. This complicates BNFL’s task considerably, and also means that in some cases BNFL must modify existing plant and structures. Such modifications may be important in their own right. The waste storage facilities at Sellafield which form the basis of this paper were constructed around 30 to 40 years ago, and so have a limited life expectancy. This has to be borne in mind - is the integrity of the structure sufficient to sustain waste retrieval over a given period of time, particularly bearing in mind the additional loads which may have to be carried? Do the loads affect significantly the seismic resistance of the plant and structure? Resistance to earthquake damage is of particular interest: BNFL has had to ensure that the new waste retrieval equipment does not unduly affect the seismic capability of the buildings which support it. This is particularly relevant to B38, in view of the considerable weight of the swarf retrieval facility. [Note that most of the waste storage facilities were not designed to resist earthquakes, although their seismic capability has been assessed retrospectively.]

vii) Off-Site Commissioning. The Sellafield waste retrieval equipment in question often has to be installed in areas of complex layout and with high background radiation, making working conditions very difficult. This necessitates remote handling, extensive use of off-site facilities for testing and training, and special attention to ALARP (as low as reasonable practicable) considerations, and calls for a thorough identification of safe working practices and effective administrative controls. I have come to regard off-site commissioning as an essential component of the safety case of a waste retrieval project. In the two examples quoted in this paper, and elsewhere, off-site commissioning has proved invaluable, both as a tool for ironing out operating problems, and as a radiologically benign means for training operators. I am aware that the operating personnel themselves appreciate the benefit of this training.

viii) Novel Plant. The use of novel plant to effect waste retrieval is often unavoidable, but it does mean that the licensee must pay more attention to design proving studies, model testing and full scale mock-ups. It also means that the regulator should pay special attention to the assessment of the adequacy of the plant.
CONCLUSIONS

This paper explains the way in which HM Nuclear Installations Inspectorate regulates projects on nuclear chemical plant, using examples of existing Magnox waste retrieval projects. I set out the key lessons learnt from regulating waste retrieval projects. Of prime concern is that Magnox sludge and swarf, which has existed for some years mainly in a form which presents a potential threat due to its physical and chemical characteristics, should be retrieved and immobilised as a matter of priority. Often it is necessary to retrieve waste from buildings which were not designed with retrieval in mind, and this requires careful consideration of the effect of the work on the existing buildings and may necessitate the use of novel plant.

HM Nuclear Installations Inspectorate has gained valuable experience through its regulation of waste retrieval projects at Sellafield. I hope that this experience will benefit members of the OECD/NEA Working Group on Fuel Cycle Safety.

FURTHER READING

In 1991, the Health and Safety Commission’s Advisory Committee on the Safety of Nuclear Installations published an excellent report on the accumulation of radioactive waste at Sellafield (Ref 3). The report expresses clearly the key safety implications of waste accumulation, and makes several recommendations concerning the direction which NII should take in regulating this work.
REFERENCES


UK CLASSIFICATION OF RADIOACTIVE WASTES

The reader should note that Magnox sludge and swarf are classed as Intermediate Level Waste, in accordance with the classification system set out by the Radioactive Waste Management Advisory Committee (RWMAC), Ref 4.

The three categories adopted by RWMAC are:

High-level or Heat Generating Wastes (HGW)

Wastes in which the temperature may rise significantly as a result of their radioactivity, so that this factor has to be taken into account in designing storage or disposal facilities.

Low-level Wastes (LLW)

Wastes containing radioactive materials other than those acceptable for dustbin disposal, but not exceeding 4 GBq/te alpha or 12 GBq/te beta/gamma.

Intermediate-Level Wastes (ILW)

Wastes with radioactivity exceeding the boundaries for low-level waste, but which do not require the generation of heat to be taken into account in the design of storage or disposal facilities.
LIST OF FIGURES

Figure 1       B30 General View
Figure 2       B38 General View
MISE EN SERVICE D'UNE INSTALLATION : L'EXEMPLE DE MELOX

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L'objet de cette présentation est de montrer, par ses aspects réglementaires et techniques, le déroulement d'une procédure d'autorisation de mise en service d'une installation nucléaire civile importante.

L'exemple choisi pour illustrer cette présentation est l'usine MELOX, implantée sur le site de Marcoule. Cette usine est destinée à fabriquer des assemblages combustibles pour réacteurs à eau sous pression, à partir d'un mélange d'oxydes d'uranium et de plutonium (MOX).
1. - Présentation de la réglementation

La sûreté nucléaire comprend l'ensemble des dispositions prises à tous les étapes de la conception, de la construction, du fonctionnement et de l'arrêt définitif des installations nucléaires, pour prévenir les accidents et, le cas échéant, en limiter les effets.

La France dispose d'un système d'organisation et de réglementation spécifique à la sûreté nucléaire. L'option fondamentale sur laquelle repose ce système est la responsabilité technique de l'exploitant. Les pouvoirs publics veillent à ce que cette responsabilité soit pleinement assumée dans le respect des prescriptions réglementaires. La définition et la mise en œuvre de la politique en matière de sûreté nucléaire est confiée à la DSIN (Direction de la Sûreté des Installations Nucléaires), relevant du ministre chargé de l'industrie et du ministre chargé de l'environnement.

Les grandes installations civiles appelées Installations Nucléaires de Base (INB), sont réglementées par le décret du 11 décembre 1963 modifié, qui définit les différentes procédures d'autorisation de création et de mise en service. Les INB sont également soumises aux prescriptions des décrets du 6 novembre 1974 et du 31 décembre 1974, qui fixent les régimes d'autorisations de rejets d'éffluents radioactifs gazeux et liquides.

Les procédures

A la suite de la demande déposée par l'exploitant, la procédure d'autorisation de création est instruite par la DSIN. Cette procédure comprend une consultation du public (enquête publique) et des autorités nationales et locales, ainsi qu'une consultation des organismes techniques (examen par le groupe permanent d'experts placés auprès de la DSIN). Un projet de décret est ensuite présenté à la Commission Interministérielle des Installations Nucléaires de Base (qui regroupe l'ensemble des ministères concernés : environnement, industrie, santé, transport, intérieur, agriculture, ...). A l'issue de cette procédure, un décret d'autorisation de création est signé par le premier ministre, sur rapport des ministres chargés de l'environnement et de l'industrie.
La mise en service des installations nucléaires de base est subordonnée à une autorisation des ministres chargés de l'environnement et de l'industrie, après examen par la DSIN et ses appuis techniques du dossier établi par l'exploitant (rapport provisoire de sûreté, règles générales d'exploitation de l'installation et plan d'urgence interne). Cette autorisation est assortie de la notification de prescriptions techniques.

De plus, une autre procédure est menée pour l'autorisation de rejets d'effluents radioactifs (liquides et gazeux). Une étude préliminaire est d'abord adressée par l'exploitant aux ministres chargés de l'industrie et de l'environnement, au plus tard lors du dépôt de la demande d'autorisation de création. Après consultation des différents ministères concernés et de l'Office de Protection contre les Rayonnements Ionisants (OPRI), une prise en considération est alors notifiée à l'exploitant. Ce dernier peut ensuite déposer une demande accompagnée d'une étude définitive. Une consultation du public (enquête publique) et des autorités locales est alors effectuée. A l'issue de cette procédure, l'autorisation de rejets est accordée par arrêté signé par les ministres chargés de l'environnement, de l'industrie et de la santé.

2. - L'usine MELOX (procédure réglementaire)

L'autorisation de création :

Le 20 novembre 1987, la COGEMA a déposé une demande d'autorisation de création de l'installation MELOX. Cette demande a fait l'objet d'une enquête publique locale qui s'est déroulée du 29 février au 31 mars 1988. A l'issue de cette enquête et de la conférence administrative réunissant les services locaux concernés, le préfet du Gard a remis un avis favorable. Un projet de décret a été présenté à la Commission Interministérielle des Installations Nucléaires de Base lors de sa séance du 16 mars 1989. La création de l'usine MELOX a été autorisée par décret en date du 21 mai 1990.

Ce décret précise que l'usine MELOX est destinée à fabriquer des crayons et des assemblages combustibles pour les réacteurs nucléaires à eau, à base d'oxydes mixtes d'uranium et de plutonium.

La capacité annuelle de production de cette installation sera de 115 tonnes d'oxyde mixte contenu dans les éléments combustibles. A aucun moment la quantité d'oxyde de plutonium présent dans l'installation n'excédera 14 tonnes.

Le plutonium mis en oeuvre ne contiendra pas plus de 3% en masse d'américium 241 et devra contenir au moins 17% d'isotope 240.

L'autorisation de rejets :


Ces arrêtés définissent les conditions et les modalités de rejets radioactifs de l’installation MELOX et imposent des limites maximales annuelles d’activité rejetée. Ces limites sont :

- en ce qui concerne les effluents gazeux :
  2 gigabecquerels pour l’ensemble des radioéléments,
  74 mégabecquerels pour l’activité totale alpha.

- en ce qui concerne les effluents liquides :
  3,3 gigabecquerels pour l’ensemble des radioéléments,
  120 mégabecquerels pour l’activité totale alpha.

Ces limites ne représentent qu’un maximum en deça duquel il y a lieu de maintenir l’activité et les quantités rejetées toujours aussi basses que possible.

La mise en service :

Le 22 août 1994, le ministre chargé de l’environnement et le ministre chargé de l’industrie ont autorisé la mise en service de l’installation MELOX.

L’installation MELOX comprend principalement deux bâtiments :

Un premier bâtiment, dit de fabrication, assure les fonctions suivantes :
- réception et entreposage des matières nucléaires (oxyde d’uranium et de plutonium), des constituants d’assemblages et des crayons MOX provenant d’autres installations,
- fabrication d’assemblages combustibles MOX,
- entreposage des assemblages.

Un deuxième bâtiment, dit bâtiment CID (Conditionnement et Incinération des Déchets), assure les fonctions :
- conditionnement des réfus de fabrication non recyclables directement en fabrication, pour expédition et traitement dans l’établissement COGEMA de La Hague,
- traitement et conditionnement des déchets technologiques : les déchets combustibles sont incinérés.

Dans un premier temps, seule la mise en actif du bâtiment de fabrication des combustibles est autorisée. La mise en actif du bâtiment CID, qui interviendra ultérieurement, reste soumise à l’approbation de la DSIN.

3.- L’usine MELOX (analyse technique)

Lors de l'analyse technique, les différents risques présents dans l'installation ont été recensés. Il faut distinguer d'une part les risques d'origine nucléaire directe :

- risque de dissémination de substances radioactives.

Ce risque, du aux matières radioactives mises en œuvre (oxyde de plutonium et d'uranium) sous différentes formes physiques, est plus important dans les unités de traitement de poudres et plus réduit lorsque celles-ci sont compactées et frittées.

La prévention contre ce risque est obtenue par la mise en place de deux systèmes de confinement, formés d'une ou de plusieurs barrières de confinement statiques, elles-même complétées par un confinement dynamique assuré par différents réseaux.

L'objectif est de maintenir en fonctionnement normal, un taux de contamination atmosphérique négligeable, à savoir inférieur au seuil de détection, dans l'ensemble des locaux. Pour maintenir ces objectifs, l'enceinte de confinement et les locaux ont été répartis en différentes classes de confinement. À chaque classe correspond un ensemble de mesures prenant en compte les risques considérés (par exemple, le nombre d'étages de filtration).

- risque d'exposition aux rayonnements ionisants.

Ce risque est présent à toutes les étapes de la fabrication. Il est principalement dû aux rayonnements bêta et gamma et aux émissions neutroniques provenant des mélanges d'oxyde d'uranium et de plutonium et des produits de filiation, notamment l'américium 241.

L'objectif retenu est de limiter, selon le principe ALARA, le nombre agents susceptible de recevoir une dose de 5 mSv/an. Cette limite correspond au dixième de la réglementation actuelle et au quart des recommandations de la CIPR 60.

- risque de criticalité.

Les caractéristiques enveloppes retenues pour l'oxyde d'uranium et l'oxyde de plutonium sont :

<table>
<thead>
<tr>
<th>Masse</th>
<th>% de Pu 239</th>
<th>% de Pu 240</th>
<th>% de Pu 241</th>
<th>% de Pu 242</th>
<th>% de U 235</th>
<th>% de U 238</th>
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<tr>
<td>Plutonium</td>
<td>&lt; 71 %</td>
<td>&gt; 17 %</td>
<td>&lt; 11 %</td>
<td>&gt; 1 %</td>
<td>&lt; 1,2 %</td>
<td>&gt; 99,8 %</td>
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<tr>
<td>Uranium</td>
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<tr>
<td>U 235</td>
<td>&lt; 1,2 %</td>
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<tr>
<td>U 238</td>
<td>&gt; 99,8 %</td>
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La prévention du risque est principalement assurée par les contrôles de masse et de modération en ce qui concerne les ateliers poudre, par la géométrie en ce qui concerne l'entreposage des pastilles, des crayons et des assemblages.

- risque du aux dégagements thermiques.
- risque lié à la radiolyse.
Il faut prendre en compte d'autre part les risques d'origine non nucléaire :

- risque d'incendie ou d'explosion.

Ce risque est à considérer en particulier dans les locaux mettant en œuvre les poudres du fait des risques radiologiques que pourrait entraîner la perte de la fonction confinement.

Un effort a donc été effectué pour diminuer la densité de charge calorifique des locaux concernés. Par ailleurs, une sectorisation des différentes parties de l'installation en Secteur de Feu et de Confinement, Secteurs de Feu et Secteurs Protégés a été mise en place.

- risque lié aux manutentions, pouvant entraîner une dissémination de matière radioactive.

- risque lié à la perte de l'alimentation électrique et du contrôle commande.

- risque associé à la perte des fluides.

- risque d'inondation d'origine interne.

- risque lié au séisme.

4. - L'usine MELOX (les inspections)

Le décret du 11 décembre 1963 définit le rôle des inspecteurs des INB. Ces derniers sont chargés de surveiller l'application de la réglementation technique générale, des dispositions contenues dans les décrets d'autorisation de création et des diverses prescriptions et autorisations.

Par les inspections, l'autorité de sûreté s'attache à vérifier que les dispositions imposées par les textes réglementaires sont respectées et que les installations sont construites et exploitées conformément aux rapports de sûreté qui les décritent.

Compte tenu de la taille et de la complexité des installations nucléaires, il est exclu que les inspecteurs examinent de façon systématique l'ensemble des locaux, des circuits et des problèmes posés par une installation nucléaire, ou en d'autres termes, qu'ils se substituent à l'exploitant. Seul un contrôle par sondage est effectué. Ils ne s'agit pas de sondages sur échantillons représentatifs, tels que peuvent l'effectuer des instituts spécialisés, mais d'examens ponctuels portant aléatoirement sur divers éléments concourant à la sûreté.

Pendant la période allant de 1990 à 1994, l'autorité de sûreté a effectué 21 visites de surveillance sur l'installation MELOX avec, comme principaux thèmes, le suivi de la construction et la mise en place des équipements importants pour la sûreté.
Conclusion:

L'ensemble des dispositions de conception, de construction et d'exploitation mises en œuvre pour assurer la prévention des différents risques, a fait l'objet d'analyses de sûreté systématiques et d'un suivi rigoureux par l'autorité de sûreté.

Au terme d'une procédure comprenant une consultation du public (enquête publique) et des autorités nationales locales, ainsi qu'une consultation des organismes techniques (examen par le groupe permanent d'experts), la mise en service de l'installation MELOX a été autorisée.
TECHNOLOGICAL ASPECTS OF SAFE OPERATION
OF REPROCESSING PLANT RT-1 IN RUSSIA

B. S. Zakharkin, E. G. Kudriavtsev, E. G. Dzekun

Abstract

Analysis of problems of extractions on the NPP's spent fuel reprocessing, related to explosion, fire and nuclear safety is made for the RT-1 plant. Factors causing the distribution and consumption of extractants are considered. The main trends of solving the problem of the plant safe operation are shown.

Introduction

The first Russian radiochemical plant on spent power fuel reprocessing RT-1 was commissioned in 1976-77. Before that there was a long period of choosing the site for the new production in a number of regions in Russia and design activities. And it was logical and justified that finally the plant was located with reference to the first object of the radiochemical industry—Production Union "Majak", located in the Urals region between the regional centres of Sverdlovsk (Ekatherinburg) and Chelyabinsk (fig.1, [1,2]). PU "Majak" had a developed infrastructure, large industrial process stock in the field of chemical technologies, and specialists experienced in a wide range of fields.

In the early 60s here the first Russian radiochemical plant ("B") on the extraction of weapon plutonium from the irradiated uranium [3] finished its operation life. The site and part of buildings of this production appeared to be suitable for the location and construction of a new plant for civil purposes.

From the start-up period the RT plant, as pilot-industrial by its first status, was subjected to modernization from time to time; its functional capabilities were being extended. To-date the functional capabilities are described by the schematic diagram shown in fig.2. The three technological chains provide for the reprocessing of a wide range of irradiated fuel of the following reactors: VVER-440; fast BN-600 and BN-350; transport and research reactors, as well as fuel elements of commercial reactors with high
uranium-235 enrichment [4]. The production rate of the VVER-440 fuel reprocessing chain makes up 400 t/year, that allows to reprocess all the fuel of this type from both Russian NPPs and those built abroad [5].

Radiochemical production, being a potential carrier of hazards, inherent in usual chemical productions (fires, explosions, toxicity), also include a specific kind of hazard - radiation effect on the human being organism in case of accident.

An accidental display of the chemical factors proper is characterized in the radiochemistry by an incomparably less probability than in such productions as, for example, the enterprises of oil chemistry, organic synthesis. However, superimposing of radiation factors on a chemical accident can cause extremely severe consequences. And the world industrial radiochemistry has vivid examples of this.

The safety problems of complicated multifunctional production, as the RT plant is, were the objects of permanent attention at all the stages of the formation and operation of the plant.

As it will be shown further, almost 20-years period of the plant operation can be assessed in terms of emergency, as highly satisfactory. This is the result of the synthesis from the previous not always satisfactory experience of radiochemical productions.

**Emergency incidents, took place at the RT-1 plant**

During 1993 the most full analysis was conducted of the statistics on accidents, emergency situations and incidents, potentially creating such situations, for radiochemical and chemical/metallurgical productions of Minatom of Russia. When processing the materials we followed the following classification:

1. Physical and physical/chemical nature of events:
   - critical incidents;
   - ignition of nuclear materials;
   - thermal and thermal/chemical processes, creating fire- and explosive situation;
   - ignitions and fires;
   - leakages of liquid and gaseous radioactive and chemically dangerous substances (corrosion damage of equipment, deviations from permissible air pressure, not leaktight stop
valves, overflow of apparatus);
- other incidents, creating emergency situation (breakages, failures of equipment, fallings down from a height, etc.).
2. Belonging to particular technological processes.
3. Organizational/engineering reasons (errors by the personnel, not working conditions of the equipment, non-qualitative designs of the processes, apparatus, incorrectly taken decisions).

This report's information is limited solely by the RT-1 plant data. The analysis is based on approximately 85 events happened during the whole period of the plant operation (1976-1993), including the start-up and adjustment period.

The generalized results are presented in fig. 3 - 5.
The followings are the conclusions caused by the analysis data:
1. No cases of nuclear materials ignition, as well as fires ever took place at the plant.
2. Among the events related to nuclear safety (fig.3) not a single case of self-supporting chain nuclear reaction has taken place. All the events of this type, amounted to 13% of the sum of the emergency situations, recorded only deviations from the specified values of the parameters. And with this it should be taken into account that the values being specified have considerable safety margins. Alongside with the events of exceeding the norms of plutonium charges, the excesses of the concentration levels being specified or plutonium charge in extractors at a long equipment operation (10-20 hours) in the "beyond-limit" mode (2 such deviations are recorded).
3. Thermal and thermal/chemical processes of the explosion character. Alongside with the passive incidents, causing a potential risk, but having no consequences (uncontrolled losses of extractant; use of HNO₃ with the concentration of more than 12 mol/l to prepare reactants; crystallization of nitrate salts in pipelines in case of exceeding the permissible ratio of aqueous solutions evaporation), the events of this type had also the real exhibitions: "puff" of the air-gas mixture in the bath of idle ends cutting-off - due to unsufficient ratio of gas phase dilution; failure of metal ceramic filter of the unit of the initial fuel solution clarification - due to the oxidizing process under the conditions of the apparatus being for a long time with the closed blow-off valve; upset of the lid of the apparatus for the preparation of hydrazine nitrate at the very
Fast change of nitric acid on hydrazine hydrate: rupture of vacuum service line of the reactor of holding the melt of hexahydrate of urea nitrate. Only the last incident among all the incidents which took place at the RT plant, can be classified according to the International scale of nuclear events (INES), with the criterion value equal to 3 ("on-site effect"). In this case an explosion took place, the reason of which was presumably the formation and accumulation of products of TBP decomposition ("red oil") in the service lines.

4. According to the first category of the emergency incidents classification (fig.3) the main number of events is caused by the leakages of radioactive and chemically aggressive substances (about 40% of all the events) and failures and breakages of the equipment (37%).

5. By the belonging to the technological process stage (fig.4), the operations related to the materials transfer (13.6%), fuel dissolution (17.3%), obtaining of the final forms of radionuclides (13.6%), waste treatment (17.3%) are characterized by the highest frequency of emergency incidents. The followings are some examples of the most considerable incidents: breakage of the crane’s suspension with the fall-off the jacket: empty or with assemblies (violation of the instruction - disconnection of the interlocks when the equipment is faulty); sticking of the fuel elements in the flow of the dissolution apparatus, releases of aerosols; aerosol contaminations of the air of the maintenance zone in case of loss of tightness by the equipment when receiving plutonium dioxide; overfilling of can for acceptance of high active glass from the furnace of electrical boiler - due to the control mechanism failure.

6. By the organizational-engineering symptoms (fig.5) in about 60% of the emergency situations the reasons for the incidents were errors, and carelessness of the personnel, violations of the operational instructions. About 30% of the incidents results from the equipment malfunctions. In 8% of cases the responsibility lies on the designs, incorrectly taken decisions, incomplete knowledge of potential risk.

7. For the whole period of the RT-1 plant operation not a single incident with a fatal result or noticeable traumas of the personnel has taken place.
Frequency situation associated with the extraction stages of preparations.

As it follows from fig. 4, a relatively small number of emergency incidents (6%) amounts for the extraction stages of the technology.

At the same time, the processes using organic systems are characterized by a high potential risk in terms of fire, explosion and nuclear safety. This potential risk finds its expression in the following:

- thermal- and radiation/chemical decomposition, oxidation of the extractant components with the formation of mixtures and adduct of organic substances with metal nitrates, forming explosive systems of thermal or detonation character;
- uncontrolled migration of extractant with aqueous solutions, accumulation of fissile substances in the organic phase being separated, with the possible fall outside the criticality boundary values;
- possible segregation of the extractant, saturated with the compounds of the fissile metals, into two organic phase with the excess of permissible metal concentration in "heavy" organic phase;
- possible in-apparatus (in the extractor) accumulation of fissile substances at the deviations in the ratios of the process flows and their composition.

In connection with this, as the priority activities, associated with the commercial application of the extraction, a great complex of investigations has been accomplished and is currently under way on the fire-risk and explosive properties of substances and mixtures, thermal, thermochemical and radiation stability of extraction systems, mechanism of oxidizing processes [6]. Results of these investigations are put as the basis for the regulation of the process parameters (limitation of the process temperature, concentrations of HNO and nitrate salts; determination of the passing ability of gas-air lines; installation of required monitoring devices, automatic equipment, safety devices, etc.).

The diluents are selected with the temperatures of burst exceeding 90° C, and having a satisfactory high radiation-chemical stability.

Many-years investigations were devoted to the theory of
extrusion emulsions. Revealing the nature of stabilizers for these emulsions and search of special substances - de-emulgators. Conclusions important for the practice followed from these investigations; concerning the methods of solutions preparation for extraction and the process conditions (operation of apparatus on the required type of emulsions, application of separating devices, emulsion de-emulgators) and etc.

Studies are carried out on the simulation of the emergency situation, related to the in-apparatus accumulation of uranium, plutonium (IV), (VI) in the double-phase extraction systems. The time parameters are set to achieve local "peak" concentrations of fissionable substances in case of deviations from the extraction technology.

A special cycle of investigation was carried out on the forecasting, synthesis and evaluation of extractants - analogues to the tributylphosphate, providing for the homogenous solution stability in carbohydrogen diluents to segregation into two organic phases at complete saturation with uranium and plutonium [7]. Such extraction systems are of particular advantage to maintain the condition of nuclear safety at the final stages of plutonium purification or when reprocessing the fast reactor's fuel. One of the specimens of such extractants, triisoamylphosphate, has passed successfully through a 3-years commercial testing at the RT-1 plant.

The balance account of the extractant is one of the important and labour-intensive tasks in the extraction process control.

It is exactly at the insufficient system of control over the organic diluent distribution that the above considered emergency situations can occur.

Factors affecting the extractant consumption are the following:

- extractant solubility in aqueous solutions;
- carry-over in the form of hardly decaying emulsions with water solutions;
- carry-over with gas phase due to volatility;
- hydrolytical and radiation/chemical destruction;
- withdrawal of the extractant during periodical conditioning of equipment off precipitates and interphase formations;
- removal of the extractant out of the cycle due to deterioration of its characteristics.

All those factors were systematically studied, and their
justification is put as a basis for the permanent system of monitoring for the extractant distribution (losses) at the RT-1 plant.

As applied to the conditions of the VVER-440 fuel reprocessing (fig. 6), the table shows a characteristic balance for TBP.

Table. Balance of TBP in the cycles of the VVER-440 fuel reprocessing, kg/t uranium (without account of the spent extractant removal).

<table>
<thead>
<tr>
<th>Process stage</th>
<th>Ways of TBP losses</th>
<th>Total losses of TBP</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Solubility! Carry-over! Radiation-</td>
<td>TBP</td>
</tr>
<tr>
<td></td>
<td>in aqueous! in the form! with gas! chemical de-</td>
<td></td>
</tr>
<tr>
<td></td>
<td>solutions! of emulsions! phase! composition</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>1 and 2 cycles of extrac-</th>
<th>6.7 ± 3.7</th>
<th>6.3 ± 2.3</th>
<th>1.0 ± 0.3</th>
<th>0.5 ± 0.4</th>
<th>15.0 ± 4.0</th>
</tr>
</thead>
<tbody>
<tr>
<td>tion</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

| Extrac- tion af- | 11.2 ± 1.0 | 2.0 ± 0.7 | 10.5 ± 0.1 | 3.6 ± 2.0 | 17.3 ± 2.4 |
|                |             |             |             |             |             |
| finage of Pu and Np|             |             |             |             |             |
| (two cyc-les)     |             |             |             |             |             |

As it can be seen from the given data, solubility and carry-over with aqueous solutions account for about 90% of all the TBP losses in 1 and 2 extraction cycles. Radiation-chemical decomposition (50%) is a determining component of TBP losses at the stage of Pu and Np affinage.

Any excess in the extractant losses caused the intervention into the technology, the reasons of deviations were found out and immediately removed.
Conclusions

The many-years experience of the RT-1 plant operation gives evidence to the satisfactory protection of the technological processes against accidents. At the same time, we suppose that the situation indicated is not a full assurance against the probable occurrence of emergency incidents.

For the enterprises of external stages of nuclear fuel cycle, to which radiochemical productions are related, it is noted that no main official document is available that specifies the general principles, classifies the safety systems and legalizes the requirements for safety analysis. The principle of "internal self-protection" in accordance with which the systems of potentially dangerous productions should have physical'chemical properties which exclude the possibility of severe accident [8], is applied insufficiently.

Further progress in providing safety in radiochemical technology is expected to be achieved as a result of R&D complex to be conducted in 1994-96 according to the accepted Industry's Programme.
References


Fig. 2. Functional capabilities of the RT plant.

Dotted line shows the operations, for which the productions required are under construction or justification (waste disposal)
Fig. 8. Classification of accidents and emergency situations (physical and physical/chemical basis of events)

- Deviations from the nuclear safety parameters
- Thermo-chemical effects
- Leaks of radioactive solutions
- Equipment failures
Fig. 4. Distribution of emergency events by the technologic symptom.
SYMPTOMS:

Errors by the personnel.
Violation of Tech Specs regulations

Equipment malfunction

Non-qualitative design.
Incomplete knowledge of potential risk. Incorrectly taken decision

Fig. 5. Distribution of emergency situation by organizational-engineering symptoms
Fig. 6. Principal scheme the extraction reprocessing of the VVER-440 fuel
La prise en compte du risque de dissémination des matières radioactives dans les usines du cycle du combustible en France

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Service d'Évaluation et de Prévention des Risques
1. OBJET DE L’EXPOSE

Dans cet exposé sont présentés les principes de conception retenus pour traiter le risque de dissémination de matières radioactives dans les usines du cycle du combustible nucléaire en France. L’ensemble des usines du cycle du combustible allant de l’enrichissement de l’uranium à la fabrication des différents combustibles et au retraitement de ces derniers après utilisation dans les centrales est concerné (usines d’enrichissement et de fabrication du combustible, usines de retraitement, installations de traitement et d’entreposage des déchets à l’exception des ouvrages de stockage). Il s’agit d’un exposé général qui ne traite toutefois pas certains cas particuliers comme la prise en compte spécifique du confinement des gaz dans les usines d’enrichissement.

La prise en compte du risque de dissémination de matières radioactives dans ce type d’installations est réalisée, dans le cadre d’une démarche de défense en profondeur, par la mise en place de systèmes de confinement successifs. Ces systèmes de confinement sont dimensionnés en fonction du risque de dissémination de matières radioactives et du risque d’exposition externe.

Dans cet exposé, nous nous attacherons à présenter :

- la réglementation, les normes, les guides,
- les différentes composantes nécessaires à la réalisation du confinement dans les usines du cycle du combustible et la façon dont elles sont réalisées,
- les évolutions des méthodes et des conceptions,
- la surveillance,
- le retour d’expérience.

Bien que les travaux de recherche ne soient pas développés dans cet exposé, on peut noter qu’ils constituent une aide importante à l’analyse et à la conception des systèmes de confinement. Un certain nombre de programmes en cours de réalisation permettent d’étudier des phénomènes particuliers tels que les transferts d’aérosols notamment en cas d’incendie, le développement de codes permettant, par exemple, d’étudier plus en détail le comportement d’un système de ventilation en situation dégradée. Une part des travaux de recherche contribue par ailleurs au développement de nouveaux matériaux (dans le domaine de la filtration par exemple).

Les principes exposés fondent les démarches des concepteurs et des exploitants nucléaires aussi bien que celles des organismes de sûreté. Rappelons que le Département d’Évaluation de Sûreté de l’Institut de Protection et de Sûreté Nucléaire a pour mission d’évaluer, à la demande de l’Autorité de Sûreté française, les dossiers présentés par les concepteurs et exploitants nucléaires dans le cadre des procédures réglementaires.
2. CRITÈRES GÉNÉRAUX DE CONCEPTION

De manière générale, les risques doivent être identifiés et pris en compte de sorte que les dispositions de conception et d’exploitation les éliminent ou rendent leurs conséquences acceptables dans le cadre des objectifs visés, tant pour le personnel que pour l’environnement. Il s’agit principalement, dans notre cas, de la prise en compte :

- des risques présentés par le procédé,

- des risques associés aux circuits de ventilation (exposition externe par accumulation de matières dans les circuits, propagation d’incendie...),

- des risques internes ou externes pouvant affecter la fonction de sûreté du confinement (incendie, perte des alimentations électriques, agressions externes...).

Les critères généraux de conception doivent prendre en compte, compte tenu des objectifs de sûreté visés, la nécessité d’assurer la permanence de la fonction confinement, celle-ci étant généralement une fonction importante pour la sûreté. Les principaux critères généraux de conception sont les suivants :

- critère de défaillance unique : redondance des éléments importants pour la sûreté, séparation physique et géographique (prise en compte des modes communs),

- spécialisation, isolement possible de portions de circuits, séparation des réseaux de ventilation des bâtiments et des procédés,

- dimensionnement au séisme de tout ou partie des barrières de confinement et, dans certains cas, des circuits de ventilation,

- classement des locaux en zones de radioprotection,

- classement des locaux en secteurs de feu ou secteur de feu et de confinement,

- surveillance flasque du fonctionnement,

- possibilité de contrôles, d’essais, d’entretien.

3. TEXTES DE BASE

En France, le dimensionnement des systèmes de confinement s’appuie sur un certain nombre de textes réglementaires, de règles fondamentales de sûreté, de normes ou de guides. La réglementation spécifique aux rejets radioactifs n’est pas présentée dans cet exposé.

3.1. TEXTES LÉGISLATIFS

Les textes réglementaires sont pour la plupart regroupés dans la brochure n° 1420 : Protection contre les rayonnements ionisants, textes législatifs et réglementaires. Les principaux sont les suivants.


Ce décret définit, notamment, la classification des personnes en fonction des doses auxquelles elles peuvent être exposées (travailleurs de catégories A ou B, personnes du public) et les limites de doses admissibles pour chacun de ces groupes. Les limites annuelles d’incorporation (LAI) et les limites dérivées de concentration des radionucléides dans l’air (LDCA) pour l’exposition professionnelle sont répertoriées dans l’annexe IV de ce décret (voir glossaire p 17).

b - Décret 75-306 du 28/04/1975 modifié par décret n° 88-662 du 06/05/1988 relatif à la protection des travailleurs contre les rayonnements ionisants dans les installations nucléaires de base.

Ce décret définit, notamment, les différents types de zones de radioprotection et leurs modes de signalisation.

c - Les arrêtés pris en application des différents articles du décret 75-306.

Un certain nombre d’arrêtés précisent les modalités d’application de différents articles du décret ci-dessus. Pour ce qui concerne le sujet traité ici, nous en citerons principalement deux qui fixent :

- les seuils et les modalités de signalisation des zones spécialement réglementées ou interdites à l’intérieur de chaque zone réglementée (arrêté du 07/07/1977),
- la périodicité des contrôles effectués pour les dispositifs de ventilation et de filtration ainsi que pour les dispositifs de surveillance (arrêté du 07/10/1977).

Zones de radioprotection

La réglementation prévoit le classement des locaux en zones de radioprotection en fonction des taux d’exposition extrême et de contamination observés dans les locaux en fonctionnement normal.

<p>| Zones spécialement réglementées ou interdites |</p>
<table>
<thead>
<tr>
<th>Zone surveillée</th>
<th>Zones contrôlées</th>
</tr>
</thead>
<tbody>
<tr>
<td>Débit de dose ambiant</td>
<td>2,5 μSv/h (0,25 mrem/h)</td>
</tr>
<tr>
<td>Contamination atmosphérique LDCA</td>
<td>1/10</td>
</tr>
<tr>
<td>Matérialisation des zones</td>
<td>(bleue)</td>
</tr>
</tbody>
</table>


3.2. REGLES FONDAMENTALES DE SURETE (RFS)

Les Règles Fondamentales de Sûreté (RFS) sont éditées par la Direction de la Sûreté des Installations Nucléaires (DSIN) qui dépend du Ministère chargé de l'industrie. À la différence des décrets et arrêtés cités ci-dessus qui doivent être strictement respectés, les exploitants peuvent ne pas appliquer une RFS s'ils apportent la preuve que les objectifs de sûreté visés par la règle sont atteints par d'autres moyens. Pour ce qui concerne le confinement des matières radioactives dans les laboratoires et usines, il en existe deux :

- la RFS II-2 série "u" - systèmes de ventilation,
- la RFS I.4.a série "u" - protection contre l'incendie.

La RFS "ventilation" expose les exigences applicables à la conception et à l'exploitation des systèmes de ventilation assurant des fonctions de sûreté dans les installations pour protéger les personnes du public et l'environnement contre les risques de dissémination de matières radioactives induits par l'exploitation de ces installations.

La RFS "incendie" expose les dispositions générales de conception et d'exploitation à prendre pour les installations à l'égard des risques d'incendie, compte tenu des risques radioactifs que ceux-ci peuvent induire.

3.3. NORMES

Les normes françaises sont éditées par l'Association Française de Normalisation (AFNOR). Comme dans le cas des RFS citées ci-dessus, les exploitants peuvent ne pas appliquer les normes. Cependant, le respect des normes en vigueur permet d'éviter de reprendre de manière détaillée l'évaluation de la sûreté des éléments concernés. Les normes applicables au confinement des laboratoires et usines sont nombreuses. On peut citer :

- les normes de la série M - "combustible et énergie nucléaire" et plus particulièrement le groupe M62 - "Installations - Enceintes de confinement",
- les normes de la série X - "normes fondamentales, normes générales" et plus particulièrement le groupe X 44 : "Filtration de l'air - Enceintes de sécurité".

Parmi les normes du groupe M62 se trouvent notamment celles qui définissent les classes d'étanchéité des enceintes de confinement et les méthodes d'essais permettant de s'assurer de la validité des étanchéités retenues.

Parmi les normes du groupe X44 se trouvent notamment celles qui définissent les méthodes de mesure de l'efficacité des filtres.

3.4. GUIDES

Il existe différents guides édités par l'Institut de Protection et de Sûreté Nucléaire (IPSN). Ces guides, sans être réglementaires, apportent un certain nombre de renseignements principalement sur les grands principes généralement appliqués et sur les matériaux existants. Ils constituent une aide à la conception. Ils sont élaborés grâce à la collaboration des analystes de sûreté, des concepteurs et des exploitants. Comme dans le cas précédent leur utilisation facilite l'évaluation de sûreté de l'installation. Concernant le confinement, il s'agit principalement :

- du guide de ventilation,
- des catalogues PMDS (Protection, Manipulation, Détection, Sécurité).

5. LES DIFFERENTES COMPOSANTES DU CONFINEMENT

5.1. GENERALITES

Dans les usines, le procédé met en œuvre des matières radioactives sous différentes formes physico-chimiques (poudres, solutions aqueuses ou organiques, gaz, ...). En fonction du procédé, ces différentes formes peuvent être présentes en même temps dans l'installation.

Le procédé nécessite l'utilisation d'installations qui peuvent être complexes : enceintes de confinement reliées entre elles, appareils chaudronnés (dissolveurs, homogénéiseurs, cuves,...), moyens de transport dans l'installation (réseaux pneumatiques, circuits de solutions ou d'éffluents, conteneurs,...), emballages de transport vers d'autres installations...

La diversité des produits mis en œuvre, la présence géographiquement étendue à des niveaux généralement significatifs de matières radioactives contribuent à la complexité de l'étude du risque de dissémination de matières radioactives, chaque installation ayant, de fait, des caractéristiques différentes.

Le confinement est obtenu par l'association du confinement statique, formé par les parois des appareils du procédé, des cellules et des locaux, et du confinement dynamique réalisé par la ventilation. Pour concevoir le confinement, il est donc très important de connaître les caractéristiques relatives au terme source (produits mis en œuvre).

La surveillance du confinement est assurée notamment par les mesures de radioprotection et les mesures réalisées sur les réseaux de ventilation (dépressions, températures,...).

5.2. CONFINEMENT STATIQUE

Le confinement statique est assuré par deux systèmes de confinement :

- le premier système, assuré par les appareils chaudronnés, les enceintes, les emballages de transport, etc., confine les matières radioactives au plus près des sources,

- le deuxième, constitué par les locaux environnants, formé d'une ou deux barrières selon l'importance des risques, renforce la protection statique vis-à-vis de l'environnement.

5.2.1. Premier système de confinement

Le rôle du premier système est d'assurer le confinement des matières radioactives au plus près de la source. Il peut être formé d'une ou deux barrières. C'est le premier système de confinement qui assure la protection du personnel vis-à-vis du risque de dissémination de matières radioactives. Il doit donc être réalisé et analysé avec soin.

Généralement, les appareils de procédé sont placés dans des enceintes de confinement dont on vérifie le niveau d'étanchéité. Il existe des exceptions à cette règle, notamment lorsqu'il s'agit d'appareils chaudronnés étanches ne présentant pas de risques d'exposition externe incompatibles avec la présence de personnel (cuves d'effluents par exemple).

Ce système de confinement comporte des traversées permettant l'introduction des matières, le passage des câbles d'alimentation électrique ou de commande, l'évacuation des déchets et effluents,... Ces traversées doivent être conçus de telle sorte que le niveau d'étanchéité requis soit reconstitué.

L'efficacité du confinement statique est éprouvée, lors de la conception, par différentes méthodes.

Il peut s'agir de contrôles en usine (par exemple le contrôle des soudures des appareils chaudronnés) ou sur site (par exemple les tests d'étanchéité des enceintes). La norme NF M 62-200 définit les classes d'étanchéité des enceintes en fonction de la toxicité des produits traités. Les normes M 62 - 210 à 213 définissent les méthodes de contrôles des taux de fuite définis.

<table>
<thead>
<tr>
<th>Classe</th>
<th>Taux de fuite horaire $T_f$ ou Taux de renouvellement horaire $T_p$</th>
<th>Exemple</th>
</tr>
</thead>
<tbody>
<tr>
<td>Classe 1</td>
<td>$T_f \leq 5 \times 10^{-4} \text{ h}^{-1}$</td>
<td>Enceinte de confinement à atmosphère contrôlée</td>
</tr>
<tr>
<td>Classe 2</td>
<td>$T_f &lt; 10^{-2} \text{ h}^{-1}$</td>
<td>Enceinte de confinement à atmosphère très dangereuse en permanence</td>
</tr>
<tr>
<td>Classe 3</td>
<td>$T_f &lt; 10^{-1} \text{ h}^{-1}$</td>
<td>Enceinte de confinement à atmosphère pouvant être très dangereuse</td>
</tr>
<tr>
<td>Classe 4</td>
<td>$T_f &lt; 1 \text{ h}^{-1}$</td>
<td>Enceinte de confinement à atmosphère pouvant devenir dangereuse en cas d'opération spéciale</td>
</tr>
<tr>
<td>Classe 5</td>
<td>$T_p &gt; 1 \text{ h}^{-1}$ avec une vitesse d'admission en tout point $&gt; 0,5 \text{ m s}^{-1}$</td>
<td>Enceinte de confinement à atmosphère à faible potentiel à risque</td>
</tr>
</tbody>
</table>

Taux de fuite ou taux de renouvellement horaire selon la norme M 62-200

La norme internationale n° ISO 10468-2 (NF M 62-230) : "Enceinte de confinement - partie 2 - classification des enceintes selon leur étanchéité et méthodes de contrôles associés" doit paraître prochainement. Cette norme fera état de 4 classes de confinement au lieu de 5, les enceintes définies

par un taux de fuite supérieur à celui de la classe 4 étant hors du domaine d'application de cette norme. Par ailleurs, à l'exception de la classe 1, les taux de fuite admis pour les classes définies dans cette norme seront plus contraignants que ceux présentés ci-dessus.

5.2.2. Deuxième système de confinement

Le deuxième système de confinement est formé par l'ensemble des locaux classés en zones surveillée ou contrôlée au sens du décret 75-306 (cf § 3.1.).

De manière générale, l'accès aux locaux abritant les enceintes proprement dites se fait par des sas matériels et personnels, le plus souvent distincts, équipés de moyens de radioprotection. Ces locaux, sauf exception, ne présentent pas de paroi en contact direct avec l'extérieur.

De même que pour les enceintes, un taux de fuite peut être exigé pour certains locaux (cas des locaux classés secteurs de feu et de confinement par exemple, voire pour certains bâtiments lorsque ceux-ci restent l'unique protection de l'environnement, après séisme par exemple). Dans ce cas, les traversées des parois sont également conçues pour respecter le niveau d'étanchéité requis.

5.3 CONFINEMENT DYNAMIQUE

Le confinement dynamique, assuré par la ventilation, complète le confinement statique. Il assure les fonctions de sûreté suivantes :

- confinement de la radioactivité, notamment en cas de défaillance du confinement statique (rupture de barrière),
- évacuation des dégagements thermiques lorsque le procédé le nécessite.

5.3.1. Prise en compte des risques engendrés par le procédé

La présentation qui suit s'appuie fortement sur le guide de ventilation édité par l'IPSN (voir § 3.4.). Initialement édité en 1982, une révision de ce guide a vu le jour en 1987.

La conception des réseaux de ventilation découle du classement des locaux en fonction du taux de contamination atmosphérique en fonctionnement normal et incidentel ou accidentel.

Pour chaque local, on définit une famille de ventilation, un niveau de dépression par rapport aux locaux et enceintes avoisinants de manière à créer une cascade de dépressions des locaux présentant le moins de risques vers les locaux à risques les plus importants et enfin les enceintes de confinement, ainsi qu'un taux de renouvellement horaire.

Les paragraphes qui suivent présentent les principes de conception des systèmes de ventilation utilisés actuellement pour des installations neuves. En réalité, il y a toujours des écarts à l'application de ces principes, écarts éventuellement importants pour les installations anciennes. Le but de l'analyse de sûreté est d'identifier ces écarts et surtout de juger de leur acceptabilité du point de vue des risques.

a - Définitions des familles de ventilation

Le tableau ci-après récapitule les définitions des différentes familles de ventilation en fonction des taux de contamination atmosphérique dans les locaux.

<table>
<thead>
<tr>
<th>Famille</th>
<th>Contamination permanente admissible (LDCA)</th>
<th>Contamination accidentelle maximale (LDCA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>IIA</td>
<td>≤ 1</td>
<td>≤ 80</td>
</tr>
<tr>
<td>IIIB</td>
<td>≤ 1</td>
<td>≤ 4 000</td>
</tr>
<tr>
<td>IIIA</td>
<td>≤ 80</td>
<td>≤ 4 000</td>
</tr>
<tr>
<td>IIIIB</td>
<td>&lt; 4 000</td>
<td>≥ 4 000</td>
</tr>
<tr>
<td>IV</td>
<td>&gt; 4 000</td>
<td>&gt; 4 000</td>
</tr>
</tbody>
</table>

LDCA : Limite Dérivée de Concentration dans l'AIR

Ce classement, datant de 1982 (et partiellement de 1987 lors de la révision du guide) tend à être remis en cause actuellement, pour être à la fois simplifié et complété. Certaines simplifications seraient tout à fait justifiées.

Par exemple, le guide présente plusieurs schémas-type de ventilation pour un même niveau de risque en tenant compte du zonage de radioprotection ce qui peut apporter des confusions dans son utilisation. Or, le risque d'exposition externe, dans les locaux et enceintes, n'intervient pas dans le dimensionnement des circuits de ventilation.

En outre, le guide de ventilation n'aborde pas la prise en compte du risque d'incendie dans la conduite de la ventilation. Ce sujet fait actuellement l'objet de réflexions dans différents groupes de travail (voir § 5.3.2. d).

Actuellement, on s'oriente donc vers l'utilisation d'un classement des locaux en "classes de confinement" plutôt qu'en "familles de ventilation", classement qui devrait permettre de prendre en compte le risque de dissémination sans ambiguïté par rapport au zonage de radioprotection.

On peut illustrer cette tendance par le tableau suivant qui reste cependant un exemple pouvant évoluer en fonction des études en cours :

<table>
<thead>
<tr>
<th>Classe de confinement des locaux et des cellules</th>
<th>C1</th>
<th>C2</th>
<th>C3</th>
<th>C4</th>
</tr>
</thead>
<tbody>
<tr>
<td>Niveau de contamination permanente (LDCA)</td>
<td>0</td>
<td>&lt;0,1</td>
<td>&lt;0,1</td>
<td>Variable</td>
</tr>
<tr>
<td>Niveau de contamination occasionnelle (LDCA)</td>
<td>0,1</td>
<td>1</td>
<td>4000</td>
<td>4000</td>
</tr>
<tr>
<td>Niveau de contamination occasionnelle (fréquence)</td>
<td>Très faible</td>
<td>Faible</td>
<td>Moyen</td>
<td>Élevé</td>
</tr>
</tbody>
</table>

Répartition des locaux en classes de confinement

b - Cascades de dépressions

La ventilation doit assurer une cascade de dépressions entre les locaux des différentes familles.

D'une manière générale, les dépressions sont de l'ordre de :
- Famille I : Pression atmosphérique ou légère supression,
- Famille II : de -8 à -10 daPa
- Famille III : de -12 à -14 daPa
- Famille IV : < -22 daPa

Dans les enceintes de famille IV, la dépression peut atteindre -50 daPa lorsque les produits manipulés sont particulièrement dangereux (plutonium par exemple).

c - Taux de renouvellement horaire

Les taux de renouvellement horaire des locaux, c'est-à-dire les débits de ventilation locaux, peuvent difficilement faire l'objet de recommandations a priori ; ils dépendent de plusieurs facteurs :
- le classement par rapport au risque de contamination et notamment le taux de contamination permanente à viser,
- l'existence éventuelle de risques spécifiques (gaz inertes, gaz inflammables ou explosifs),
- le maintien du confinement dynamique en situation incidentelle ou accidentelle, qui dépend par exemple de la taille des ouvertures postulées de la barrière de confinement,
- le conditionnement climatique (dégagements thermiques à évacuer).

En pratique, les taux de renouvellement horaires s'échelonnent, pour un certain nombre d'installations typiques, de 1 à 5 environ pour des locaux correspondant respectivement aux familles I à IV.

d - Dispositifs de filtration et d'épuration

Les dispositifs de filtration et d'épuration en fonction des familles de ventilation sont présentés dans les tableaux suivants.

Les principes appliqués pour l'élaboration de ces tableaux sont essentiellement les suivants :
- éviter le transfert de la contamination dans d'autres locaux ou enceintes par la mise en place de filtres à l'extraction mais également au soufflage lorsque c'est nécessaire,
- mise en place d'une filtration au plus près de la source d'émission potentielle de matières radioactives afin de limiter l'importance des volumes contaminés,
- assurer une redondance des moyens de filtration en fonction du risque pour pallier la défaillance d'un niveau de filtration et protéger l'environnement en toutes circonstances,
- assurer la permanence de la ventilation, et de la filtration notamment pour le dernier niveau de filtration à l'égard de l'environnement, pendant les interventions sur les filtres.

Des systèmes de contrôle et de surveillance des transferts de contamination sont indiqués sur ces tableaux. Ceux-ci peuvent être renforcés ou adaptés en fonction de l'installation.

Par ailleurs, un certain nombre de dispositifs permettent de s'assurer de la permanence de la filtration, notamment la dépression aux bomes des filtres pour suivre l'évolution du colmatage qui peut ensuite être compensé, dans un premier temps par le réglage des registres prévus à cet effet puis par le remplacement des filtres.

5.3.2. Prise en compte des risques engendrés par la ventilation

Les principaux risques qui peuvent être engendrés, propagés, ou aggravés par la ventilation sont les suivants :

- exposition aux rayonnements,
- dissémination de matières radioactives,
- incendie.

Pour prévenir ces risques, un certain nombre de mesures sont prises en compte au cours des études de conception. Ces risques sont prévenus par les dispositions suivantes :

a - Risque d'exposition externe

La prévention de ce risque est réalisée en évitant les dépôts de matière dans les circuits par :

- l'étude des tracés des circuits : absence de "bras morts", rayons de courbure adéquats, mise en place d'éléments filtrants au plus près des sources, vitesses d'écoulement adaptées,...
- la nature des circuits : acier inoxydable, acier noir peint,...

Lorsque le dépôt de matière ne peut pas être évité (filtre de premier niveau, par exemple) des protections biologiques sont mises en place.

b - Risque de dissémination

Pour prévenir ce risque, on prend les dispositions suivantes :

- éviter le recyclage des rejets de l'installation ou d'une installation voisine (choix de l'emplACEMENT des prises d’air, hauteur des cheminées,...),
- choix d'un niveau d'étanchéité approprié des gaines notamment jusqu'au premier étage de filtration,
- maintien de la permanence de l'extraction pendant le changement des premiers niveaux de filtration,
- maintien de la permanence de la filtration pendant le changement du dernier niveau de filtration,
### Tableau III : CLASSEMENT PAR FAMILLES

<table>
<thead>
<tr>
<th>RISQUES DE CONTAMINATION</th>
<th>ORGANISATION DE LA VENTILATION EN FONCTION DES RISQUES DE CONTAMINATION</th>
<th>EXEMPLES</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>FAMILLE I</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CP :</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CA :</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 1, FI</td>
<td></td>
<td>Zone 1</td>
</tr>
<tr>
<td>Recyclage</td>
<td>Centrale de filtration</td>
<td>Salle de conduite</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 1 ou 2, FI</td>
<td>Zone 2</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 1 ou 2 ou 3 ou 4, FI</td>
<td>Zone 3</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 1 ou 2, FI</td>
<td>Zone 2</td>
<td>THE(S)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 2 ou 3 ou 4, PIE</td>
<td>Zone 3</td>
<td>THE(S)</td>
</tr>
<tr>
<td>Transfert de zone 1 ou 2 ou 3, FI</td>
<td>THE(S)</td>
<td></td>
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<td></td>
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<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 2 ou 3 ou 4, PIE</td>
<td>Zone 4</td>
<td>THE(S)</td>
</tr>
<tr>
<td>Transfert de zone 1 ou 2 ou 3 ou 4, FI</td>
<td>THE(S)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>FAMILLE II A</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CP S1 LOCA</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CA ≥ 80 LOCA</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 2 ou 3, PIE</td>
<td>Zone 3</td>
<td>THE(S)</td>
</tr>
<tr>
<td>Transfert de zone 1 ou 2 ou 3, FI</td>
<td>THE(S)</td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 2 ou 3 ou 4, PIE</td>
<td>Zone 4</td>
<td>THE(S)</td>
</tr>
<tr>
<td>Transfert de zone 1 ou 2 ou 3 ou 4, FI</td>
<td>THE(S)</td>
<td></td>
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<tr>
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<td></td>
</tr>
<tr>
<td><strong>FAMILLE II B</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CP S1 LOCA</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CA ≤ 4000 LOCA</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 3, PEB</td>
<td>Zone 3</td>
<td>THE(S)</td>
</tr>
<tr>
<td>Transfert de zone 2 ou 3 ou 4, PEB</td>
<td>THE(S)</td>
<td></td>
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<td></td>
</tr>
<tr>
<td>Soufflage ou transfert de zone 3 ou 4, PEB</td>
<td>Zone 4</td>
<td>THE(S)</td>
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<tr>
<td><strong>EXEMPLES</strong></td>
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<tr>
<td>ZONE : 1</td>
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<td></td>
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<tr>
<td>Salle de conduite</td>
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</tr>
<tr>
<td>ZONE : 2, 3, 4</td>
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</tr>
<tr>
<td>Coulées</td>
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<td>Zones à intervention</td>
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<tr>
<td>Laboratoires</td>
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<tr>
<td>Salle refroidissement</td>
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<tr>
<td>ZONE : 3, 4</td>
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</tr>
<tr>
<td>Evaporateurs P.F.</td>
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<tr>
<td>Cellules chimiques fraudés</td>
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<tr>
<td>Traitement d'eau</td>
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<tr>
<td>Stockage résines</td>
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<tr>
<td>Traitement pouvant</td>
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<tr>
<td>Désinfecteurs</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Stockage Produits de fissi</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**OCDE - Cadarache 20/21 septembre 1994 - La prise en compte du risque de dissémination de matières radioactives dans les usines du combustible en France.**
### Tableau III : CLASSEMENT PAR FAMILLES (Suite)

<table>
<thead>
<tr>
<th>RISQUES DE CONTAMINATION</th>
<th>ORGANISATION DE LA VENTILATION EN FONCTION DES RISQUES DE CONTAMINATION</th>
<th>EXEMPTES</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>FAMILLE III A</strong>&lt;br&gt;C.P. ≤ 80 L.D.C.R&lt;br&gt;C.A. ≤ 4000 L.D.C.R</td>
<td>[Diagramme 1] Soufflage ou transfert&lt;br&gt;de zone 3, F.E.A ou zone 3, F.E.A&lt;br&gt;Possibilité de prélèvement&lt;br&gt;Zone 3&lt;br&gt;HEIS_3&lt;br&gt;THEIS_3&lt;br&gt;THEIS_3&lt;br&gt;Transfert de zone 2 ou 3, F.E.A&lt;br&gt;</td>
<td>[Diagramme 2] Soufflage ou transfert&lt;br&gt;de zone 3 ou 4, F.E.B ou zone 3 ou 4, F.E.A&lt;br&gt;Possibilité de prélèvement&lt;br&gt;Zone 4&lt;br&gt;HEIS_3&lt;br&gt;THEIS_3&lt;br&gt;THEIS_3&lt;br&gt;</td>
</tr>
<tr>
<td><strong>FAMILLE III B</strong>&lt;br&gt;C.P. ≤ 4000 L.D.C.R&lt;br&gt;C.A. importante&lt;br&gt;plus de 4000 L.D.C.R</td>
<td>[Diagramme 3] Soufflage ou transfert&lt;br&gt;de zone 2 ou 3, F.E.A ou EB, zone 3, F.E.A&lt;br&gt;Possibilité de prélèvement&lt;br&gt;Zone 4&lt;br&gt;THEIS_3&lt;br&gt;THEIS_3&lt;br&gt;</td>
<td>[Diagramme 4] Soufflage ou transfert&lt;br&gt;de zone 2 ou 3, F.E.A ou FEB&lt;br&gt;Possibilité de prélèvement&lt;br&gt;Zone 1&lt;br&gt;THEIS_3&lt;br&gt;THEIS_3&lt;br&gt;</td>
</tr>
<tr>
<td><strong>FAMILLE IV</strong>&lt;br&gt;C.P. de l'ordre de 4000 L.D.C.R&lt;br&gt;C.A. très faible&lt;br&gt;plus de 4000 L.D.C.R</td>
<td>[Diagramme 5] Soufflage ou transfert&lt;br&gt;de zone 2 ou 3, F.E.A ou FEB&lt;br&gt;Possibilité de prélèvement&lt;br&gt;Zone 6&lt;br&gt;THEIS_3&lt;br&gt;THEIS_3&lt;br&gt;</td>
<td>[Diagramme 6] Soufflage ou transfert&lt;br&gt;transfert spécifique&lt;br&gt;Précédé&lt;br&gt;Zone 6&lt;br&gt;THEIS_3&lt;br&gt;THEIS_3&lt;br&gt;</td>
</tr>
</tbody>
</table>

- **Prélèvement in situ**
- **Détection (β)**
- **Contrôle (γ)**
  - Filtre haute efficacité
  - Filtre très haute efficacité
  - CP : Contamination permanente
  - CA : Contamination accidentelle
  - S_1 : Systèmes permettant la filtration et l'extraction d'air en permanence
  - S_2 : Systèmes permettant l'extraction d'air en permanence, pendant le remplacement de la filtration


274
- changement des éléments filtrants sous protections étanches,
- montage en dépression des éléments filtrants (en amont des ventilateurs et non en aval),
- choix des débits permettant d’assurer des vitesses d’écoulement d’air adéquates dans certains cas d’ouverture du premier système de confinement (ouverture de sas d’introduction de matériels) ou rupture du confinement statique (arrachage d’un gant d’une boîte à gants).

c - Séisme

Selon les conclusions de l’analyse de sûreté relative au risque dû au séisme, tout ou partie des circuits de ventilation peuvent être dimensionnés au séisme en fonction du but recherché :

- éviter que des éléments importants pour la sûreté ne soient détruits par des projectiles provenant de composants des installations de ventilation qui pourraient se décrocher au cours d’un séisme,
- conserver les circuits en état après séisme,
- disposer d’une installation de ventilation en état de fonctionnement après séisme.

d - Incendie

En ce qui concerne les circuits de ventilation proprement dits, ceux-ci doivent être dimensionnés pour ne pas propager un incendie d’un local vers un autre.

Les locaux sont classés en Secteurs de Feu (SF) ou Secteurs de Feu et Confinement (SFC) dont on peut rappeler les définitions ci-après :

Secteur de Feu : on appelle secteur de feu chaque volume élémentaire délimité par des éléments de construction dont le degré de résistance a été choisi en fonction de l’incendie considéré comme plausible qui s’y déclarerait ou le menacerait et les moyens de secours programmés pour le temps correspondant au degré de résistance qualifiant ce secteur de feu.


La ventilation est alors dimensionnée en tenant compte des critères suivants :

- possibilité d’arrêter et de fermer le soufflage par des clapets coupe-feu (CCF) pour éviter l’apport d’air neuf alimentant l’incendie,
- possibilité de fermer l’extraction également par des CCF pour protéger la dernière barrière de filtration,
- réalisation de conduits résistants en température,
- possibilité de contournement du premier niveau de filtration par un bipasse étanche pour éviter son colmatage trop rapide et sa destruction.

- dilution des gaz chauds par de l'air frais provenant de locaux présentant un même niveau de risque (respect de la séparation des circuits) en amont du dernier étage de filtration pour en assurer la protection tout en permettant de conserver l'extraction dans le local en feu,

Il n'y a pas de règle générale concernant la conduite de la ventilation d'un SFC, la règle essentielle étant la protection de la dernière barrière de filtration, celle-ci étant l'ultime protection de l'environnement. La conduite peut être difficile en fonction de la complexité des réseaux. Chaque installation fait l'objet d'une analyse détaillée pour élaborer les procédures de conduite les mieux adaptées. Il est à noter que ce type de préoccupation fait actuellement l'objet de réflexions au sein de groupes de travail et est supporté par un programme important de recherche et développement.

Le schéma suivant présente un exemple simplifié de schéma de ventilation de secteurs de feu isolables séparément :

6. SURVEILLANCE DU CONFINEMENT

La surveillance des systèmes de confinement est réalisée par la mise en place d'un réseau d'appareils de radioprotection.

Pour ce faire, d'une façon très générale, on peut noter les règles suivantes.

Un certain nombre de contrôles peuvent être réalisés dans les gaines de ventilation en fonction des besoins : prélèvement, détection bêta, contrôle gamma.

Les locaux abritant des enceintes de confinement sont équipés de moyens fixes de mesures permanentes de la contamination et de l’irradiation ambiante délivrant des alarmes en fonction des seuils atteints.

Les sas d’accès à ces locaux sont équipés de boîtiers d’accès permettant de connaître les niveaux de contamination et d’irradiation dans le local en fonction des seuils retenus ainsi que de moyens fixes et mobiles de contrôles du personnel sortant.

Les rejets à la cheminée sont surveillés en permanence (prélèvement sur filtre des aérosols, mesures en continu pour les cheminées rejetant l’air en provenance des cellules, des enceintes et des appareils de procédé...).

7. RETOUR D’EXPERIENCE

Au niveau du retour d’expérience, des modifications peuvent être apportées aux installations suite aux évolutions de celles-ci (modification du procédé, des produits traités,...) et des mises à niveau susceptibles d’être introduites à la suite du développement de nouveaux matériaux (R & D) ou pour pallier des problèmes que le retour d’expérience aurait fait apparaître au cours de la vie des installations.

Les enseignements du retour d’expérience ont essentiellement porté sur :

- la séparation des réseaux de ventilation en fonction des risques présentés par les différents locaux desservis. Les locaux de famille I sont raccordés à un réseau spécifique. A chaque famille de ventilation correspond un réseau d’extraction particulier.

- la mise en place de filtres sur des circuits de transfert d’air de locaux vers des enceintes,

- la limitation voire la suppression, autant que faire se peut, des ventilateurs intermédiaires, rendant plus facile le réglage des dépressions en cas de défaillance de l’un d’entre-eux.
8. GLOSSAIRE

AFNOR : Association Française de Normalisation
CCF : Clapet Coupe-Feu
CEA : Commissariat à l'Energie Atomique
DSIN : Direction de la Sûreté des installations Nucléaires
IPSN : Institut de Protection et de Sûreté Nucléaire

LAI : Limite Annuelle d'incorporation par ingestion ou inhalation : pour un radionucléide donné, activité incorporée en un an dont la valeur est la plus faible des deux valeurs suivantes :

a - celle qui entraîne un équivalent de dose engagé égal à 0,5 Sv pour l'organe ou le tissu le plus irradié,

b - celle qui entraîne la valeur de 0,05 Sv pour la somme des équivalents de dose engagés, au niveau des différents organes ou tissus, pondérés par des coefficients appropriés.

LDCA : limite dérivée de concentration d'un radionucléide dans l'air : concentration moyenne annuelle dans l'air inhalé, exprimé en unité d'activité par unité de volume qui, pour 2 000 heures de travail par an, entraîne une incorporation égale à la limite annuelle d'incorporation par inhalation ou, pour les gaz rares autre que le radon, entraîne un équivalent de dose égal à l'une des limites annuelles d'exposition fixées par l'article 9 du décret 86-450 du 20/06/1986.


RFS : Règle fondamentale de sûreté
R & D : Recherche et développement
SF : Secteur de Feu
SFC : Secteur de feu et de Confinement

Personnel de catégorie A : travailleurs directement affectés à des travaux sous rayonnements : personnes dont les conditions habituelles de travail sont susceptibles d'entraîner le dépassement des trois dixièmes des limites annuelles d'exposition fixées aux articles 9, 10 et 11 du décret 86-450 du 20/06/1986.

Personnel de catégorie B : travailleurs non directement affectés à des travaux sous rayonnements : personnes dont les conditions habituelles de travail sont telles qu'elles ne peuvent normalement pas entraîner le dépassement des trois dixièmes des limites annuelles d'exposition fixées aux articles 9, 10 et 11 du décret 86-450 du 20/06/1986.
SAFETY ASSESSMENT OF FIRE POSTULATED IN A PROCESS CELL
OF FUEL REPROCESSING PLANT

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Abstract

Organic solvent fire in a process cell is one of the postulated events for examination of safety design of fuel reprocessing plant. NUPEC/JINS, under the auspices of Science and Technology Agency, have developed computer codes to be used for assessing the safety of plant through the fire event, including their validations, and made the assessment analysis for the licensing examination of the Rokkasho Fuel Reprocessing Plant. The flow and temperature distributions along the ventilation systems including the process cell where the organic solvent was burning were evaluated. The analysis showed that, because of decrease of the burning rate in the cell isolated, of heat transfer to the cell walls, and of mixing of the massive outlet flows from other cells, the air temperature flowing into the filter unit as well as the pressure loss due to accumulated soot, would not exceed the allowable limit. Then, integrity of the radioactivity containment systems was assured through the event. However, the computed results also suggested that the heat transfer from the flame to the cell walls should be examined more precisely. For this purpose, two dimensional air flow and temperature distribution analysis with the MAC method was made. Comparative studies with temperature distributions observed in cell fire and ventilation experiments indicated that a practical heat transfer model was valid in thermofluid analysis of the cell fire, and that a precise heat transmission modeling for thermal barriers covered cell walls was useful for prediction of the temperature distributions. Through the safety analysis for the licensing examination, also indicated was strong dependence of the burning rate of the solvent on the oxygen content in the cell or the ventilation rate, and related phenomena of the solvent burning will be studied in detail.

1. Introduction

In the Regulatory Guide for Licensing of Reprocessing Plant, prepared by the Japanese Nuclear Safety Commission, organic solvent fire in a process cell is listed as one of the Design Basis Events should be examined in evaluation of safety design of reprocess-
ing plant. To support establishing criteria necessary for the evaluation, the Science and Technology Agency conducted a series of safety proving tests in the 1980s: Cell fire and ventilation experiment, HEPA filter test under accidental severe condition and so on. In the cell fire and ventilation experiment, burning characteristics of organic solvent in pool fire and radioactive material release from burning solvent were observed,\textsuperscript{(1,2)} and FACE computer code based on the results of the experiment was developed.\textsuperscript{(3)} In the HEPA filter test, allowable limits of temperature and soot accumulation load of the filter were obtained.\textsuperscript{(4)}

At the Institute of Nuclear Safety, study of the solvent fire in a process cell of reprocessing plant was started in 1986, aiming at an application of the FACE code to the evaluation of the design basis event of fire for licensing the Rokkasho reprocessing plant. Having made modifications and validation tests of the FACE code, the evaluation of the fire event was completed in 1992 and the safety design of the plant was confirmed.\textsuperscript{(5)} Through the study, characteristics of the solvent fire in a isolated space have been also made clear, and it was suggested that the heat transmission from fire flame to the cell walls and the burning rate of the solvent are critical factors in defining the temperature in the cell and the exhaust. Based on these findings, a new phase of the study has been started last year.

2. Preparation of FACE Code at INS

2.1 Bases of FACE Code

The FACE code developed by Nishio of JAERI has two key options: Total network analysis of the ventilation systems including a cell where fire is assumed, and the two dimensional analysis of the distribution of air flows and temperatures in the fired cell. Thermo-fluid behaviors in the network are analyzed with the node-and-junction models, and
the computing methods of the EVENT84 code\(^{(0)}\) are employed. The momentum equations for incompressible fluid, the mass equation and the energy equation with the perfect gas assumptions. Additionally prepared are the analytical models for heat transmission to the cell and ventilation duct walls, transportation of soot and radioactive materials and operations of components such as blowers or dampers.

To accomplish the evaluation of the design basis fire event, the option for the total network analysis was selected and its computing capabilities were extensively examined at INS. Numerical calculations of the heat transmission to the duct surfaces were carefully examined, because the discrete node-and-junction models of the network would bring restrictions on the junction length and the time-step width required for stable numerical integrations. It was also confirmed that the restrictions based on the criteria for stable heat transmission calculations were applicable in the soot transportation calculations, because the analogy of the heat and mass transfers was employed in the code.

In the FACE code, the transportation of the radioactive materials are treated in two categories, non-gaseous and gaseous, or condensable and non-condensable. The non-gaseous condensable radioactive materials assume to transfer and deposit, and to be removed by filter unit with soot. But, the gaseous non-condensable materials flow through the ventilation systems without any interference. These assumptions and analytical models were generally accepted in our study. Models for the operation of blower and damper, including check damper, are necessary and important for the evaluation of design basis events. Junctions for leak paths must be included in the network modeling.

Analytical model and empirical parameters for determining the burning rate of organic solvent were also prepared in the FACE code by Nishio. The parameters were developed from observations of organic solvent burnings in the cell fire and ventilation experiment. But, this model was not examined nor used in the safety evaluation activities.
at INS because it seemed not to be adequate for general applications.

After the precise verification and the modifications at INS, the code for the total network analysis of the ventilation systems was named FACE-VENT.

2.2 Validation Tests of FACE-VENT Code

The validation work of the FACE-VENT code at INS had started with an analysis of explosion problem, and the explosion sample problem presented in the Nuclear Fuel Cycle Facility Accident Analysis Handbook was selected for this study. A sketch of the ventilation system is shown in Fig. 1. This system consists of a large main room (270 m³) with a glovebox (2.4 m³), inlet supply and exhaust blowers, and filters. The model for this system is shown in Fig. 2. As described in the Handbook, the initial starting flow rates are 270 m³/min through the main flow system with 13.5 m³/min pulled through the glovebox line. With the assumption of the explosion of acetone of 50 ml, the shape of the mass and energy injections are also given in the Handbook. Based on these descriptions and conditions, the input data for the FACE-VENT code were prepared, and the transients of pressures and temperatures in the ventilation system were calculated as shown Fig. 3. These results were compared with the results of EXPAC code presented in the Handbook. Both the FACE-VENT and EXPAC codes are based on the calculation method of the EVENT84 code, so that good agreement between the results of these codes were justified. With this explosion problem, the models for damper or check damper responses were also examined.

The analysis of fire in the glovebox was studied next. Using a ventilation system similar to the system used in the explosion sample problem, the models for the heat transmission to room and duct surfaces were carefully examined. As mentioned in the previous section, when the discrete node-and-junction modeling is applied, the length of
junction with heat transmission must be shorter than the allowable maximum length deduced from the criteria for stable numerical integration. So that, the network model of the ventilation system was modified and constructed as shown in Fig. 4. Shorter junctions are prepared for the ventilation ducts in which large temperature gradients along the air flow are expected. The filter unit at the glovebox inlet is replaced with the check damper to control backward flow into the room. Typical temperature transients in the glovebox and its outlet line are shown in Fig. 5. A rapid pressure rise after the ignition causes the check damper to shut. But shortly, continuous blower suction and not-so-large heat addition assumed in this case make the glovebox pressure lower and the air supply resumes. So that, the temperature in the glovebox is maintained constant after the initial transient. A number of parametric studies indicated that these transients were dependent on the heat generation by fire, the heat loss by heat transmission, and the suction power of blower. This fire problem was also analyzed with the FIRAC code developed at LANL, and qualitatively good agreement of the time transients between the FACE-VENT and FIRAC codes was concluded.

3. Analysis of Fire Event in Rokkasho Plant

The evaluation of the design basis fire event of the Rokkasho reprocessing plant was made with the FACE-VENT code. In the application for designation of reprocessing business submitted to the Government, the postulated fire event is described as follows: Leakage of organic solvent would occur in a cell for extraction columns in plutonium purification facility, and a small amount of the solvent not recovered and left at cell bottom might be heated up to ignite by some cause or other. Radioactive materials would be released through the cell exhaust line and the ventilation line of building common area where contamination might extend.
Having taken into account these postulations and anticipated phenomena, a network model of the cell and building ventilation systems in the plutonium purification facility was prepared as shown in Fig. 6. Loss of external electric power supply is assumed simultaneously with the fire, but the exhaust blowers will continue their operations at reduced power with electricity supply from emergency generators. Sets of active and passive dampers are equipped at the cell inlet lines and they will prevent backward flows to the common areas in the building. But, in this analysis, only the passive check dampers assumes to be effective in limiting the flows from the cells. The air supply shut-off damper at the building inlet line is closed. Leaks through the cell and building walls are additionally included for conservativeness.

Heat and soot generations by solvent burning were defined by evaporation rate and burning fraction of the solvent. The evaporation rate of solvent in pool fire was observed through the fire and ventilation experiment\(^{(06)}\) and its maximum value (0.07 kg/m\(^2\)/sec) was used in this analysis. The burning fraction of the evaporated solvent depends on oxygen concentration or ventilation condition in a cell or isolated space, and in general, the fraction will decrease considerably in the insufficient oxygen condition after the initial complete burning phase. Heat transmission to the cell walls, convective and radiative, was only considered, and heat loss with flows in the ventilation ducts was ignored for conservative analysis. Convective heat transfer coefficient generally measured during fire in room or building was used (8 kcal/m\(^2\)/hr). No deposition nor removal of soot was assumed in the cell and the ventilation ducts for greater soot accumulation to the filter units.

Pressure and temperature transients calculated with the FACE-VENT code are shown in Fig. 7. Pressure rise and highest temperature in the cell are not so great, and no damage of cell integrity is anticipated. The calculated results also showed that, because of
decrease of the burning rate in the cell isolated, of heat transfer to the cell walls, and of mixing of the massive outlet flows from other cells, the temperature rise at the filter unit was not so large. A more conservative analysis, in which the complete burning of evaporated solvent was assumed, was additionally made to obtain the maximum possible amount of soot accumulated to the filter units. Then, it was confirmed that the temperature and the pressure loss due to accumulated soot did not exceed the allowable limits for HEPA filter (200 C and 3.9 kPa) established by the HEPA filter test. So that, the integrity of the radioactivity containment systems were assured through the postulated fire event, and the release of radioactive materials was limited to an very low level.

To contain and control contaminated air, the pressure inside nuclear facility building must be kept negative, or lower than the outside atmosphere, during anticipated or postulated events. Evaluation of the postulated fire event to this requirement was made with a simplified and conservative model of the building ventilation system and the fire: The heat release from the solvent burning was assumed directly in the common area in the building. And, it was clearly indicated that the pressure in the building was kept negative through the fire event.

4. Extended Studies

In the FACE-VENT code, as mentioned in the previous sections, the phenomena of cell fire are treated with the one-point model and the constant convective heat transfer coefficient is used through the transients. To improve the model and to develop a practical heat transfer coefficient applicable to the cell fire analysis, simulation studies of fire in isolated space have been conducted at INS. In its first step, two dimensional marker-and-cell method simulation code FACE-CELL, based on the two dimensional analysis option of the FACE code, was prepared and simulation of the cell fire and
ventilation experiment was made with it. Figure 8 shows one of the results obtained with
the practical heat transfer coefficient proposed and used in the FIRIN code. Stratified
temperature distributions observed in the experiment is reproduced in the simulation, but
with somewhat higher values. Through the simulation study, it was also indicated that a
heat transmission model to consider thermal barrier plates or cell liners which cover the
wall structure with gaps greatly improved the simulation results.

The burning fraction of evaporated solvent was not evaluated in the FACE-VENT
code. But, it will be easily concluded that the burning fraction depends on oxygen
concentration and ventilation rate of a isolated space. A study of developing an analytical
model to determine the fraction and to estimate its value for cell fire has been just started
at INS.

5. Concluding Remarks

Studies made at INS for the evaluation of the postulated fire event in reprocessing
plant were summarized: The FACE-VENT code for analyzing the transient phenomena in
ventilation and a fired cell systems, including pressures, temperatures, air flows and soot
transportations, was prepared, and its validation studies were conducted. Using the
FACE-VENT code, the safety analysis of the postulated fire event of the Rokkasho
reprocessing plant was carried out, and the safety design of the plant was confirmed. The
extended studies for defining the heat transmission characteristics of fire and the burning
fraction of evaporated solvent in a isolated space are going on.

Acknowledgment

This work was carried out under the auspices of the Science and Technology Agency
of Japan. The authors are greatly indebted to Messrs. Y. Nitta and S. Hosoda of the CRC
Research Institute for their considerable efforts in the verification and validation works of
the FACE-VENT and FACE-CELL codes.

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Fig. 1  Sketch of the Explosion Sample Problem Ventilation System  
(From NUREG-1320)

Fig. 2  Network Schematic of the Explosion Sample Problem Ventilation  
System  
(From NUREG-1320)
Fig. 3 Transients in Explosion Calculated with FACE-VENT Code
Fig. 4 Network Schematic of Ventilation System for Fire Problem

FIRE IN GLOVEBOX: [CALT=1 FINE NODE F088]

Fig. 5 Temperature Transients in Glovebox
Fig. 6 Network Schematic of Ventilation Systems for Evaluation of Fire Event in Process Cell
Fig. 7 Transients in Postulated Fire Event Calculated with FACE-VENT Code
VELOCITY VECTOR  MAX = 3.14E0
TIME  = 2.00E3  CYCLE NUMBER  = 4000
FFF RUN FP-8  IN CELL / FP8-C-07L

(a) Velocity Distributions

CONTOUR OF TEMPERATURE (C) TO = 2.00E1
MIN. = 2.52E1  MAX = 5.99E2  INTERVAL = 5.73E1
TIME  = 2.00E3  CYCLE NUMBER  = 4000
FFF RUN FP-8  IN CELL / FP8-C-07L

(b) Temperature Distributions with Measured Values in Parentheses

Fig. 6 Analytical Results with FACE-CELL Code

294
Moderately dense concentration - Criticality safety

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A project sponsored by the Swedish Nuclear Power Inspectorate

Summary

The title of the paper is a play with words. It refers to the important variables moderation, density and concentration. It will be shown that these variables are often used incorrectly, both in Sweden and in other countries. The mistakes cannot be attributed to any single organization or country. Neither are they limited in time; the problem exists today and has existed for at least 20 years. The conclusion is that better methods and education are required to handle moderation correctly. The problems discussed may not always be realistic. However, if they are not analysed at all, nobody will know. If emergency response is based on incorrect information, an incident can turn into an accident.

I. Introduction

Criticality safety has been an established sector in the nuclear industry since the beginning. Better data and better computer codes are now available to more people than ever.

However, the methods used for criticality safety analysis do not seem to reflect this long experience. After more than 20 years experience with criticality safety (including safety reviews and research), it is clear to me that a substantial improvement of the methods used for the analysis is possible and needed. Hundreds of people have criticality safety as their principal work. Many specialists, who have contributed so much in the past, are still active. They have a knowledge and an experience (wisdom) that will never be duplicated. Today, and in the future, we cannot expect the criticality safety sector to attract the same status, funding and dedicated specialists as in the past.

A guide is needed to reduce the difficulties in determining the influences of varying densities and concentrations. Today, many of the uncertainties are covered by extremely conservative (pessimistic) approximations. However, if not all types of incident scenarios are analysed correctly, there is no guarantee that those approximations will dominate and lead to conservative results.

II. Density or concentration - A problem?

Density is here defined as a measure of mass per geometric unit. Volume, surface and linear densities are all useful concepts in assessing criticality safety. If gU stands for grams of uranium, examples of the mentioned types of densities are gU/cm³, gU/cm² and gU/cm.

Concentration is defined as a ratio between two variables. With this definition, density is a type of concentration. Some examples are atomic ratio H/U, volume ratio H₂O/UF₄, weight-ratio (enrichment) U-235 to total uranium and the number of apples per orange in a fruit salad. All these concentrations are simple, in the sense that they compare variables of the same type (measurement unit). A more complicated concentration is the mass of uranium per volume of a homogeneous and saturated mixture of only water and uranium dioxide powder at a certain temperature and pressure. The measurement unit (incomplete) could be gU/cm³. In this case the uranium density will have the same value and the units appear to be the same. This will be discussed in section IV of the paper.

Consider the following scenario. Homogeneous ashes containing uranium and other materials have been obtained from burning contaminated filters, clothes and other burnable materials. An incident, where water can be mixed with the ashes, has to be considered. The uranium concentration refers to a homogeneous mixture of uranium powder and water at room temperature.

There are two options, could any or both of them be correct as a basis for criticality analysis?

- The ashes have a uranium concentration of 0.20 g/cm³. The uranium density is not specified.
- The ashes have a uranium density of 0.20 g/cm³. The uranium concentration is not specified.

Again, this question will be answered later in the paper (sections V.1 and VI). Surprisingly, density and concentration are often mixed up in safety reports. This can lead to serious errors in the conclusions of the safety analysis.
III. Reviews of criticality safety

The nuclear fuel cycle in Sweden is not a complete cycle. The following comments are related to operations in the fuel cycle that have been reviewed in the past. Later, some typical problems that have been found during the safety reviews will be discussed. They will be related to density, concentration and moderation.

Some errors have been found in Swedish safety reports, while others were found in reports from organisations in other countries. All reports were directly related to the Swedish nuclear fuel cycle.

III.1. Uranium mining

There is a lot of uranium in Sweden. In one place, Ranstad, a full scale mining operation was prepared but stopped for political and economical reasons. One building, with equipment originally designed for extracting uranium from the mine, today is used to purify uranium from fuel fabrication side streams.

III.2. UF₆ conversion to UO₂ and fuel fabrication

A "wet" process for conversion of uranium hexafluoride (UF₆) to uranium dioxide (UO₂) is used. This means that good moderation often is a normal parameter, not an incident. There are also many different fissile materials. Some are over-moderated even when dry. Moisture measurements will not detect moderation in such a case. AUC (ammonium uranyle carbonate) is such a material.

Large filters, accumulations of smaller filters, contaminated and other mixed materials contain hundreds of kilograms of uranium. Recovery of side streams containing uranium lead to similar quantities of uranium with low densities. Some of the materials are dry, some liquid and some in undefined states. Incidents, involving significant amounts of uranium leaking into the ground outdoors, have also occurred.

Very large storage areas, where water distribution and density are important parameters, can be found at the plant. Moderation control is a standard procedure in several areas and containers.

III.3. Storage of fresh and spent fuel

The fuel assemblies in a reactor form a lattice of closely coupled units with minimal neutron leakage. In storage areas, a margin to criticality is achieved by absorption of neutrons in the materials between the assemblies and/or by neutron leakage. Low density water and heterogeneous distribution of water is important in the safety analysis of fresh fuel.

If the burnup of the fuel is used for criticality control, consideration must be taken of the fact that the densities of U-235 and other nuclides vary in the axial direction of the fuel. This effect is varying with the time after shutdown of the reactor (radioactive decay). Care must be taken to assure that the methods are applicable to this situation.

III.4. Reprocessing and MOX fuel fabrication

The reprocessing was only laboratory scale. In connection with this activity, there were serious plans for MOX (mixed uranium/plutonium oxide) fabrication in significant quantities. An application for such an operation was reviewed 20 years ago. The operation was never carried out.

III.5. Final storage of spent fuel

Several studies have been carried out concerning final storage of spent fuel. A purpose has been to demonstrate that criticality will not be a probable event, even in a time frame of millions of years. Some of these studies have been required before starting new power reactors and have been reviewed and approved by the authority.

Sooner or later, the containers and the fuel itself will disintegrate. Exactly how this will happen is not necessary to know. When it could happen is the question. Important parameters are varying densities, burnup and radioactive decay.

III.6. Transportation of fissile materials

Reviews of safety reports for transport packages give the authorities in different countries an opportunity to examine methods used by organisations and authorities in other countries.

Unirradiated and waste uranium materials are of special interest. Those materials are not very radioactive (essentially no requirements for containment and shielding). For some reason, unknown to me, both uranium and water are often assumed to be distributed only homogeneously inside the packages.

297
IV. Criticality handbooks

In the past, criticality handbooks, guides and standards have been important as references. Most of the handbooks are now quite old. An exception is a Japanese handbook, which I have not yet seen.

Criticality handbooks include data related to the concentration of a fissile material to some mixture.

Before using such data, it is necessary to know the limitations. There are uncertainties caused by the evaluation of experiments and calculations. The values are only applicable to a certain material or nuclide related to another specified material, nuclide or mixture. But also temperature, pressure and other parameters may influence the validity of the data.

To show how different the parameters density and concentration are, consider temperature effects. The concentration of hydrogen in pure water is the same at room temperature as in ice formed by the same water at a temperature below the freezing point. The numerical values are different, about 0.11 gH/cm³ at room temperature and about 0.10 gH/cm³ in ice. But the measurement units are also different when written out completely.

0.11 [gH/cm³H₂O]ₚₑ = 0.10 [gH/cm³H₂O]ᵢₑ

However, the hydrogen densities for the two temperatures are different.

For an unlimited geometry in all three dimensions, the neutron multiplication factor for a homogenous fissile material will not change by a change of density. The concentration stays the same and will determine the multiplication factor.

For a limited geometry, the density is important. A reduced density will increase the neutron leakage from the fissile unit. This will usually reduce the multiplication, but not always. Moderation effects can change the situation, especially when interaction with other fissile units is possible. A reduced fissile density may allow other nuclides to enter the fissile material. Compression of materials with low densities must be considered.

A system with varying fissile densities can have a smaller critical mass than a homogeneous fissile system.

V. Safety reviews - Some experiences

The confusion between concentration and density is not new. Many times, applicants have had to revise their analyses to get approvals. The following description of the case with "dry water" is typical, but also extreme. It is very recent (not yet settled).

V.1. Dry water

This case is based on the ashes mentioned in section II. The U²³⁵ enrichment is 5% by weight. The applicant claimed that no restriction is required for the amounts and geometry of the material. The uranium concentration is restricted to 0.20 g/cm³. Concentration of uranium related to what?

The safety reports give the answer to this question. The concentration is such as used in a criticality handbook and in one place it is also defined as

\[ c(H₂O) = \rho(H₂O)ₚₑ (1 - \frac{c(U)}{\rho(H₂O)ₚₑ}) \]

where "c" stands for concentration, "p" stands for density and "th" for theoretical. The theoretical densities are given as 0.9982 g/cm³ for water and as 18.9 g/cm³ for uranium. This formula does not take the oxygen of uranium oxide into account, leading to a requirement for more water.

For the maximum uranium concentration of 0.20 g/cm³, the water concentration is at least 0.9876 g/cm³. For smaller uranium concentrations, the water concentration is even higher. The requirement can also be described as an atomic ratio H/U of no less than 125. Does this agree with the allowed contents - ashes? That is, essentially no water at all.

The safety reports claim that during normal operation there is no problem since the material is
dry. The atomic ratio H/U is specified as less than 0.3. Criticality is not possible for this atomic ratio.

The safety reports also claim that during accident conditions there is again no problem since the uranium concentration is less than 0.20 g/cm³. As shown above, such material is basically water (99 % by volume). Again, this makes criticality impossible.

The safety reports have used density as if it was a concentration. This is very dangerous, since there is no safe density in the same way as there are safe concentrations.

A more realistic specification for the material is ashes with a maximum uranium density of 0.20 g/cm³. There are no restrictions on other nuclides in the material. Carbon and oxygen would be expected normally in ashes. Hydrogen is not excluded (no measurements) and may be added in the form of water during an incident.

In section VI a proposal for a correct safety analysis is given.

V.2. Water - Here, there and back again
This example covers lumps of fuel separated by air, water, water spray and maybe snow or ice. Storage and transport of fuel rods, assemblies and pellets in boxes or on trays where water can remain are typical cases.

Safety analysis for transport of fresh fuel assemblies is usually based on water at reduced density only. Water at full density, filling parts of the containers is a completely different scenario. Freezing of the water (after a fire in a cold climate) could lead to geometries not credible with running water.

The normal situation is to have some plastic materials like polyethylene and ethafoam inside the package and often inside the fuel assemblies.

The packages are dry during normal transport. The fire test required by the transport regulations will often lead to damages that will allow water to flow into the damaged package.

Before the package becomes completely flooded with water, there are other alternatives. Partial drainage of the water and redistribution of the water during handling are other possibilities. They have to be evaluated.

The actual design, testing, manufacturing and handling of the package must take the water distribution during and after the tests into account.

Array of damaged BWR or pellet containers

The instructions for handling the fuel assemblies may take into account some non-homogeneous water distribution. This is obvious from the requirements for polyethylene bags around the fuel to be open at both ends. However, those requirements only cover some of the potential consequences of water in-leakage and distribution.

V.3. Surface density
The special hand calculation methods involving surface density control are not used so often anymore. But surface density control will always be an important method for criticality control.
Examples of surface density control is the restriction of storage units to one layer and limiting the slab thickness of a fissile material.

One safety report, that was reviewed some time ago, used surface density control. The fissile materials were distributed over a room, but most of the fissile materials were concentrated to one corner.

Acceptable surface densities for the various fissile materials were established. This seemed acceptable.

Large room with various fissile materials

However, the implementation of the control was not approved. The total amount of fissile materials and the floor area of the room were estimated. The ratio between the results were used as the maximum surface density for the room. In the corner of the room, with most of the fissile materials, the real surface density was of course much higher.

Really disturbing about this review was that the safety analysis had been made by an experienced criticality specialist, normally using very conservative approaches to criticality safety.

VI. Low uranium density - Analysis

Ashes, with a uranium density of not more than 0.20 g/cm³ and with varying densities of hydrogen, carbon and oxygen are the fissile materials to be analysed. The U-235 enrichment is 5% by weight. The materials are assumed to be stored in cylindrical steel containers. Inner diameter 56 cm, inner height 90 cm and steel thickness 0.3 cm. The containers can be stored in any configuration, including at a triangular pitch.

A correct safety analysis must take optimum presence of hydrogen and carbon into account. The low uranium density and the influence of carbon make the analysis complicated. The optimum is not only influenced by moderation and absorption of neutrons in the ashes, but also by neutron leakage.

A preliminary study will be reported here. It will be based on the same fissile material (ashes) as mentioned earlier. The geometry is simplified. The geometry is either infinite or a single water reflected sphere with a steel wall, 0.3 cm thick. The radius of the sphere will vary and is shown in the figures. This study will be applicable to many bulk quantities of materials with low uranium densities.

The SCALE 4 computer code package was used for the calculations (sequence CSAS1X with the 27 group burnup library).

The densities of the materials are either given in g/cm³ or fractions of their theoretical densities (10.96 g/cm³ for UO₂, 2.3 g/cm³ for C and 1.0 g/cm³ for H₂O).

The figures are shown as examples of typical parametric studies that may be included in a safety analysis.

With a fixed uranium density of 0.2 g/cm³ (corresponding to a fraction 0.0207 of the theoretical UO₂ density), the water density is allowed to vary. The sum of the fractions will be less than 1.0 which means that there are voids in the material.
The peaks do not occur at the same water density. Here, the concept "optimum moderation" is used for infinite geometry only. For other geometries, "optimum water density" is preferred. For a higher neutron leakage (smaller radius), the optimum is moved to a higher water density. The additional water is acting as an "internal reflector".

A determination of optimum water density is only valid for a certain geometry. If the geometry is changed, the optimum is changed as well. The curves are often quite flat near the optimum densities. This simplifies the analysis.

These curves have simple shapes even at the very high or low water densities no shown. For heterogeneous, low enriched fissile materials, the curves can be more complicated. A well known case is new fuel assemblies in water of varying densities.

A mixture of uranium dioxide, carbon and water seems to be more dangerous than any other mixture. This is also a reasonable scenario. The flat curves also show that the danger is not restricted to a narrow range of densities. Note that the sum of the fractions of UO₂, H₂O and C is 1.0. No voids.

The following figure shows that, when neutron leakage is considered, also oxygen (O) together with water will lead to higher neutron multiplication than only water. A curve for only water and uranium dioxide can be found in a previous figure.

![Graph showing the influence of oxygen and silicon](image)

This figure involves carbon at a very high density (close to the theoretical density 2.3 g/cm³). It will not be possible with waste materials. However, any amount of carbon will reduce the critical mass. The next figure will show results for mixtures of uranium dioxide, carbon and water. The major influence of carbon in these mixtures is as an internal neutron reflector and not so much as a moderator.

Finally, a figure showing the influence of oxygen and silicon (Si) is shown below. Even silicon may increase the neutron multiplication a little bit. The effect is not so dramatic as for carbon and also much smaller than for oxygen.

For oxygen, the curve for the infinite geometry shows that the neutron multiplication is being reduced by adding more oxygen. However, when a limited geometry is studied (here a water reflected sphere), the opposite conclusion can be drawn. It is not unusual that conclusions in safety reports are based on infinite systems and then applied to limited systems.

![Graph showing the influence of oxygen and silicon](image)
VII. Criticality achieved by nature

Criticality is not a technical process, invented by man. It is a naturally occurring process. This has been proven in Oklo, Gabon, Africa. Many million years ago, the U-235 concentration (enrichment) in uranium was higher. The mixture of uranium, water and other materials was the right for some very long critical excursions. This historical event is interesting in many ways. It demonstrates the effects of density, concentration and moderation.

VIII. Conclusions

The question of moderation is probably the most important and complicated issue in the criticality safety field.

Experiments are necessary to establish the characteristics of various materials, geometries, calculation techniques or combinations of these. They can be used to test that the optimum moderation in a specific situation can be predicted by calculations or other estimations.

However, to guide specialists in the criticality field in the complicated problems involving moderation, experiments are not the solution. More realistic, engineering models of the physics of neutron multiplication is required. The current praxis, often based on sophisticated experiments, is not enough.

A proposal is to stop using the concept "optimum moderation" when "optimum, homogeneous water distribution" is intended. Optimum moderation maybe could be restricted to unlimited geometries. In limited geometries, where neutron leakage is important, additional materials may increase the neutron multiplication. If the additional material is water, it should be considered as an "internal reflector", not as a moderator. By separating the effects of internal moderation and internal reflection, it will be more obvious for the analyst to look for other materials than water that may give internal reflection.

Similar ideas can be used for intermediate and external effects. The concepts of internal, intermediate and external reflection/moderation are combinations that will help the analyst.

Some of the safety problems discussed in the paper may not be very realistic. But, if some scenarios are not discussed at all in the safety reports, the worst should be expected. Incidents that are worse than the design criteria happen. If the approved safety reports claim that optimum moderation has been covered, when this is not true, the emergency response during an incident may make the situation critical.

IX. References

Topical meeting on safety of the nuclear fuel cycle

Study on Some Problems for Criticality Safety Evaluation in Japan

Yoshitaka NAITO and Hiroshi OKUNO

Department of Fuel Cycle Safety Research,
Tokai Research Establishment,
Japan Atomic Energy Research Institute
Criticality safety evaluation methods have been studied in Japan. Some results obtained through this study are different from those presented in the European and American handbooks or guides. The following three items of them are presented in this report. (1) The nuclear criticality conditions and their lower limits obtained from the estimated criticality neutron multiplication factors and their allowances for the computer code system JACS developed at JAERI, are a little different from those described in the handbooks of the other countries. (2) Fuel particle size obtained here for low enriched uranium dioxide capable of being regarded as homogeneous in nuclear safety analysis is different from that shown in the French criticality guide CEA-R3114. (3) The formula introduced here for the concrete wall thickness for isolating neutron is more than twice as thick as 30cm recommended by TID-7016. Finally, we propose the international cooperation to solve the above problems and to compile the International Criticality Safety Handbook.
1. Introduction

The nuclear criticality safety handbook of Japan was published by Nuclear Material Regulation Division of Science and Technology Agency of Japan in 1988. Since 1989, the study has been performed for revising the Handbook under the support of many scientists of Japan especially of JAERI. Some results obtained through this study are different from those found in the European and American handbooks. The following three items of them will be presented and discussed.

(1) Estimation method of critical lower limit.

(2) Fuel particle size capable of being regarded as homogeneous in nuclear criticality safety analysis.

(3) Concrete wall thickness for isolating neutron.

2. Estimation method for critical lower limit

In many handbooks, the criticality conditions, such as minimum critical mass, critical diameter of infinit circular cylinder and so on, and the lower limit of them are described. But the way how to obtain them, especially the lower limit, are not clear. Criticality conditions are, sometimes, obtained with computer codes but the accuracy of the code is rarely shown. The lower limits of them are, sometimes, obtained by producing safety factors which are difficult to understand how to obtain them. In the Japanese Handbook, the criticality conditions and their lower limits are obtained systematically with the computer code system JACS which were evaluated by analysing more than 1000 cases criticality experiments. As shown in Fig.1, estimated $K_{e,r}, \bar{K}$, and estimated lower limit $K_{e,l}, K_c$, for each fuel type are obtained by many benchmark calculations. A system is assumed to be critical if the calculated $K_{e,r}$ of the system by the code is $\bar{K}$ and assumed never to be critical if the calculated $K_{e,r}$ is less than $K_c$. The Estimated Criticality Neutron Multiplication Factors and the Estimated Criticality Lower Limit Neutron Multiplication Factors are obtained and shown in TABLE 1 for each fuel.
material by the computer code JACS which was developed in JAERI. With these multiplication factors, we obtained the critical diameter of an infinitely long cylinder containing homogeneous $^{239}$Pu-H$_2$O and compared them with those written in the European and American handbooks $^{19}$ - $^{21}$ as shown in Fig. 2. The values of the European and American handbooks in Fig. 2 are critical ones, not lower limit ones. This figure shows our estimated values are not so different from the European and American values. If we compare and discuss about these values internationally, it may be able to obtain critical values and their allowances which will be agreed internationally. Another example for comparison of minimum critical values is shown in TABLE 2 presented by Dr. B. Gmäi of GRS at the criticality working group meeting of OECD/NEA held in July 1994, who also proposes to obtain international criticality values.

3. Fuel particle size capable of being regarded as homogeneous in nuclear criticality safety analysis

Powdered fuel in water seems to be a heterogeneous system at the point of neutron reactivity. When the fuel particle size is, however, very small, the system may be treated as a homogeneous system. To obtain the limit of this particle size, the neutron multiplication factor was calculated for an infinite cubic array of slightly enriched UO$_2$ sphere particles immersed in water with various enrichments, water to fuel ratios and fuel particle sizes. The calculations were performed with a computer code module based on the collision probability method to solve the ultra-fine energy group (about 70,000 gr.) equations of neutrons. The change in the neutron multiplication factor from the homogeneous system is dominated by the change in the resonance escape probability, $\Delta p/p$, not in the thermal utilization factor, $\Delta f/f$, as shown in Fig. 3. The change depends almost completely on the uranium concentration and rarely on uranium enrichment up to 10 wt% for a particle size of 1 mm as shown in Fig. 4. The dependence determines the fuel particle size regarded as homogeneous in proportion to the negligible relative error of the neutron
multiplication factors. As an example, for array of 2.5 wt% -enriched UO₂ particles of 1 mm in diameter immersed in water, whose uranium concentration is 2.6 gU/cm³, is read as 1.2%. When 0.3% ΔK/K is assumed for the enough small reactivity to be negligible, the corresponding diameter is

\[
1\text{[mm]} \times \frac{0.3[\%]}{1.2[\%]} = 0.25\text{[mm]},
\]

which leads to the fact that the fuel particles, the sizes of which are no larger than 250 μm, can be regarded as homogeneous in nuclear criticality safety analyses.

In the French criticality guide "", mean free paths of thermal neutron , λ, in uranium fuel and plutonium fuel systems shown in TABLE 3, are used to determine the fuel particle sizes of heterogeneous that can be identified as homogenous.

Our present work shows that the fuel particle size regarded as homogeneous is affected by the change not in the thermal utilization factor but in the resonance escape probability. The difference between the two should be examined.

4. Concrete wall thickness for isolating neutron

Concrete wall thickness for isolating neutron was discussed. The thickness of 30cm was recommended in TID-7016 "", which was obtained from the experimental data. The experiment seems not to be general but specific. We have studied to obtain general form to determine isolation thickness.

We introduced the Reflector Factor, RF, defined by the following equation,

\[
RF = \left| \frac{k(R)-k_0}{k_0} \right|
\]

where,

K(R) : Neutron effective multiplication factor of the system where isolator thickness is R and the nuclear fuel systems on the opposite sides of the isolator are assumed to be the same.

K₀ : K(R→ ∞ ), which is the neutron effective multiplication
factor where the subject nuclear fuel system is surrounded by infinitely thick isolator.

The trend of RFs of a concrete isolator and a water isolator was examined to many kinds of fuel materials. Figure 5 shows the results of detailed calculations for the changes in RF where the three types of infinite slab fuel are isolated with concrete or water. As shown in this figure, the gradient of RF does not depend on fuel material but depends on isolator material. The gradient is approximately the inverse of the neutron migration length in the isolator except in junction area. With the above relation, we introduce the following equation to calculate isolation thickness, \( L \), simply.

\[
RF = \frac{k(0)-k_s}{k_s} \cdot \exp\left(-\frac{R-M}{M}\right) < \varepsilon
\]

\[
L = \left[ \frac{1}{\ln \left( \frac{(k(0)-k_s) / (k_s \cdot \varepsilon)}{1+\ln \left( \frac{(k(0)-k_s) / (k_s \cdot \varepsilon)}{1} \right)} \right)} \right]
\]

Where

\( k(0) : k(R=0) \), which is the effective multiplication factor of the system where the isolator thickness becomes zero and the nuclear fuel systems are in contact with each other.

\( R \) : thickness of the isolator,
\( M \) : migration length of neutron in isolator.

If we apply the following values to the above equation, very wide isolation thickness, 82.6 cm, is calculated.

\( k_s = 0.95 \) --- allowable limiting neutron multiplication factor in a single unit,

\( k(0)=k \infty = 3.0 \) --- highest infinite multiplication factor assumed,

\( \varepsilon = 0.3\% \Delta k/k \) --- cut off reactivity to neglect small one,

\( M = 10.9 \text{ cm} \) --- neutron migration length in concrete

The value 82.6 cm is too much different from 30 cm which is recommended in many handbooks or guides in the world. We are now discussing about the reason and what isolation thickness should be used. To discuss the above matter, it is important to consider the following items,
(1) An area of the room where is fuel is finite, not infinite,
(2) Number of rooms is not infinite and so k(0) is not same as k ∞ .
(3) Shape of fuel is not always slab geometry and solid angle may be taken into
account to determine isolation thickness.

5. Concluding remarks

Criticality conditions such as critical mass, critical diameter of
infinitely long cylinder, etc. are physical constants and only one value for
each condition is enough in the world. Computer codes and nuclear constant files
to obtain the criticality conditions are in progress now. It is good timing to
obtain them by international cooperation. Through this cooperation, it is
expected to be able to obtain not only criticality conditions but also thier
allowances.

Particle size which regard as homogeneous in criticality safety
analysis, isolation thickness of each material for isolating neutron, and so on
should be discussed internationally and the reasonable recommended values should
be obtained.

If the above international cooperation will be performed and the fruits
will be compiled into "International Critical Safety Handbook", the Handbook
will contribute to safety control of nuclear fuel in 21 century in the world.

References
(1) Compiled by Sci. and Tec. Agency of Japan, "Nuclear Criticality Safety
(2) Katakura, J. et al, "Developement of the Computer Code System JACS for
<table>
<thead>
<tr>
<th>Name of Group</th>
<th>Estimated Criticality Value</th>
<th>Neutron Multiplication Factor</th>
<th>Benchmark Calculation</th>
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<tr>
<td>Low Barich U</td>
<td>0.985</td>
<td>0.988</td>
<td>1.010</td>
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<tr>
<td>-Po</td>
<td></td>
<td>0.980</td>
<td>1.013</td>
</tr>
<tr>
<td>-VOX</td>
<td></td>
<td>0.990</td>
<td>1.045</td>
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<tr>
<td>High Barich U</td>
<td>0.991</td>
<td>0.994</td>
<td>1.013</td>
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<tr>
<td>-Po</td>
<td></td>
<td>0.980</td>
<td>1.010</td>
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<tr>
<td>-VOX</td>
<td></td>
<td>0.990</td>
<td>1.045</td>
</tr>
<tr>
<td>Bet. Low Barich U</td>
<td>0.985</td>
<td>0.988</td>
<td>0.984</td>
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<tr>
<td>-Po</td>
<td></td>
<td>0.980</td>
<td>0.984</td>
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<tr>
<td>-VOX</td>
<td></td>
<td>0.987</td>
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The Value of 0.992 obtained statistically is replaced by this value 0.990

*1 UX means UO₃(NO₃)₂ - Pu(NO₃)₄ system

*2 Estimated Criticality Values and their lower limits.
### TABLE 2
Comparison of minimum critical values: Spherical mass of homogeneous mixtures [kg]

<table>
<thead>
<tr>
<th></th>
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<tr>
<td>U-235 - Metal 30 cm H₂O-Reflector</td>
<td>20.1</td>
<td>21.85 / 22.8</td>
<td>23.8 $^3$</td>
<td>-</td>
<td>20.7</td>
<td>21.8</td>
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<td>U-235 - Water</td>
<td>0.76 $^5$</td>
<td>0.82</td>
<td>0.87 $^4$</td>
<td>0.88 $^4$</td>
<td>0.77</td>
<td>0.80</td>
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<td>U (20 %) O₂ - Water</td>
<td>-</td>
<td>-</td>
<td>5.2</td>
<td>-</td>
<td>5.1</td>
<td>5.0</td>
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<td>U (5 %) O₂ - Water</td>
<td>32.8 $^5$</td>
<td>-</td>
<td>35.4</td>
<td>37</td>
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<td>U-235 - Nitrate</td>
<td>0.78</td>
<td>0.85</td>
<td>0.87</td>
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<td>Pu-239 - Metal H₂O-Reflector</td>
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<td>0.547</td>
<td>0.52</td>
<td>0.49</td>
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$^5$ UO₂F₂
$^2$ ANS 8.15-1981
$^3$ U (93.5 %) 20 cm H₂O-Reflector
$^4$ U (93 %)
$^5$ SCALE 3.1

Proposal of a benchmark calculation of minimum critical values by B. Gaal and W. Weber of GRS at the criticality working group meeting of OECD/NEA held in 1994.
TABLE 3 λ-values being proportional to mean free paths of thermal neutron in uranium and plutonium fuel for determining fuel particle size (Lmm) capable of being regarded as homogeneous in nuclear criticality safety analysis with following relation,

\[ L(\text{mm}) = \frac{18.7}{\rho} \cdot \lambda (\text{mm}), \quad \rho: \text{Fuel density} \]

<table>
<thead>
<tr>
<th>Enrichment of $^{235}\text{U} (%)$</th>
<th>Pulutonium</th>
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</thead>
<tbody>
<tr>
<td>93.5</td>
<td>50</td>
</tr>
<tr>
<td>(\lambda) (mm)</td>
<td>0.06</td>
</tr>
</tbody>
</table>

(from CEA-R3114)
Fig. 1-1 Frequency distribution of the Computed results on criticality experiments by JACS
Fig. 2-1 Comparison of the Values critical diameter of an infinite cylinder containing homogeneous $^{239}$Pu-H$_2$O with those described in foreign handbooks.
Fig. 2-2  Critical masses of a bare sphere of $^{235}\text{U} - \text{H}_2\text{O}$
Fig. 3 Relative changes in the infinite multiplication factor and its four factors for arrays of 5wt% $^{235}$U-enriched UO$_2$ particles immersed in water.
Fig. 4 A summary of relative changes in $p$ and $f$ for low ($\leq 10\text{wt\%}$) enriched UO$_2$ particles (dia. 1mm) immersed in water.
Fig. 5 Change in the reflector factors vs. slab thickness of water and concrete

- ○ $^{233}$U metal (thickness 2 cm)
- • Homogeneous $^{233}$U–H$_2$O (Uranium density 0.1g/cm$^2$, thickness 7.5 cm)
- △ Homogeneous–U–H$_2$O ($^{233}$U enrichment 5 wt%, Uranium density 2g/cm$^2$, thickness 12 cm)
- $M_w$ Neutron migration length in Water (5.93 cm)
- $M_c$ Neutron migration length in Concrete
Fig. A.1 Calculation flow in JACS code system

Safety Research Programs in NUCEF for Fuel Cycle Back–end Facilities

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ABSTRACT

In JAERI–Tokai, NUCEF (the Nuclear Fuel Cycle Safety Engineering Research Facility) has just been completed in its construction and the cold functional tests are underway towards commissioning the hot operation at the beginning of 1995. The research programs which will be conducted in NUCEF are summarized by the research group in JAERI.

In this paper purpose and outline of the NUCEF, and related safety research program to criticality, confinement of radioactivities, incident and postulated accidents are shown connecting with their safety criteria and its technical bases. The planning and prospect are also shown including international cooperation as NUCEF95.

1. INTRODUCTION

Japan has a basic national policy to recycle nuclear fuel used for the effective utilization of nuclear energy and for the stable energy supply with the consideration of global environment. In harmony with the policy, JAERI (the Japan Atomic Energy Research Institute) has just completed in the construction of new facility, named NUCEF (the Nuclear Fuel Cycle Safety Engineering Research Facility)[1], towards commissioning the hot operation at the beginning of 1995. The researches in this facility aim to contribute to the safety establishment and development of advanced technologies in fuel cycle back–end.


In the former specialist meeting on safety and risk assessment in fuel cycle facilities (1991,
Tokyo), outline of the NUCEF facility had already been reported[5]. Afterwards the construction and safety inspections of the NUCEF have well been advanced. The research program is summarized for 1994 through 1998 by the NUCEF Research Program Coordinating Group which were formed in JAERI-Tokai. The managing system including the cooperation with other organization is also discussed.

In this paper the progress of NUCEF project, safety research program and prospect are shown connecting with such major engineering safety issues as criticality, confinement of radioactivities, postulated accident, especially in the reprocessing facility.

2. SAFETY RESEARCH PROGRAM IN NUCEF

As shown in the Long-term Program which has recently been renewed in Japan, the recycling of nuclear fuels is necessary from the view of effective utilization of natural and nuclear resources and conservation of global environment on the earth. Although these recycling programs are steadily performed in harmony with former program, the research and developments for the fuel cycle back-end will be pushed forwards in order to attain further safety establishment and development of advanced technologies for the next century.

Along the Long-term program, the research program in NUCEF focuses on the improvement and evaluation of nuclear criticality safety of fissile solution systems, advanced fuel reprocessing processes including partitioning of TRU (Transuranium) elements, the safety management of TRU waste, the related basic chemistry on TRU and key technologies for TRU handling. In addition, experimentally proving idea to produce advanced technologies on these subjects is another important scheme of this research program.

Three major research program, criticality safety research, research on advanced reprocessing process and TRU waste management are planned in the NUCEF. From 1991 internal review has been started on the original research themes in each program so as to maximize their effectiveness.

Table 1 summarizes the research subjects of NUCEF project. For the safety assessment of fuel reprocessing facility, the safety guide for licensing[6] has been set forth. Major safety issues for the public are confinement of radioactivities, prevention and assessment of postulated accidents like criticality. In NUCEF, criticality safety researches for preventing criticality and for assessment of postulated criticality excursion will be conducted. Furthermore, upon confinement of radioactivities, safety researches for normal operation and for postulated criticality accidents will be carried out. Those researches aim at the clarification of safety margin, the accumulation of basic data concerning the present plant. The researches will also develop both assessment methodology for postulated criticality accident and advanced measuring technologies such as sub-criticality measurement and TRU contents measurement in the waste for the future plant.
Table 1 Major research subjects in NUCEF project

<table>
<thead>
<tr>
<th>CRITICALITY SAFETY RESEARCH[*]</th>
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<tbody>
<tr>
<td>STACY (Static Criticality Experiment Facility)</td>
<td></td>
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<tr>
<td>- Obtaining basic criticality benchmark data on low-enriched U and Pu nitrate aqueous solution</td>
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<tr>
<td>- Establishment of more reasonable criticality safety margin</td>
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<tr>
<td>TRACY (Transient Criticality Experiment Facility)</td>
<td></td>
</tr>
<tr>
<td>- Obtaining transient behavior and source-term data on low-enriched U nitrate aqueous solution</td>
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<tr>
<td>- Establishment of more appropriate scenario and confinement of radioactivities for postulated criticality accident</td>
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<tr>
<th>RESEARCH ON ADVANCED REPROCESSING PROCESS</th>
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<td>Improvement of PUREX process[*]</td>
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<tr>
<td>- Obtaining basic data for safety evaluation, optimization and simplification</td>
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</tr>
<tr>
<td>- Confinement of radioactivity such as iodine and minor actinides</td>
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<tr>
<td>- Recovery and separation of TRU</td>
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<tr>
<td>Development of partitioning process</td>
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<tr>
<td>- DIDPA separation of high-level liquid waste</td>
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<td>- Development of advanced technology to separate specific nuclides</td>
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<thead>
<tr>
<th>RESEARCH ON TRU WASTE MANAGEMENT</th>
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<tr>
<td>Development of TRU waste treatment and disposal method</td>
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<tr>
<td>- Obtaining basic data on migration of TRU and clarifying its mechanism under geological condition</td>
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<tr>
<td>- Development of ceramic solidification technique</td>
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<td>- Evaluation of disposal safety</td>
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<tr>
<td>Development of TRU measuring technique[*]</td>
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<tr>
<td>- Improvement of passive/active neutron detection method</td>
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<td>- Application of gamma-ray CT (Computed Tomography)</td>
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<th>KEY TECHNOLOGY DEVELOPMENT</th>
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<tr>
<td>- Technology development of operation, maintenance and inspection of the facility[*]</td>
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<td>- Simulation technology development[*]</td>
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<td>- Development of safeguards technology</td>
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<tr>
<td>- Development of advanced treatment technology of TRU waste</td>
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</table>

[*]: on fuel cycle engineering safety

3. EXPECTED CONTRIBUTION TO SAFETY ESTABLISHMENT

(1) Safety regulation for reprocessing plants

Figure 1 shows the outline scheme of the safety regulation for reprocessing plants in Japan. According to the Regulation Laws for the nuclear facilities and Radiation control and related rule, safety review is performed by both the Science and Technology Agency (STA) and Nuclear Safety Commission. Design approval and operation inspection are conducted by STA. The Atomic Energy Commission reviews, also, the adaptation of the reprocessing business to the national policy and feasibility of financial base.
Figure 1  Scheme of the safety regulation for reprocessing plants in Japan

(2) Safety assessment

As shown above, the safety researches to be carried out in NUCEF are expected to contribute not only to safety technology improvement for the future, but also to safety assessment in the technology of current fuel cycle back-end. Major fields of the contribution in the engineering safety of reprocessing facilities are upon criticality safety, confinement process safety in a reprocessing, and TRU measurement in the waste. Expected contribution to safety assessments by the experiment in NUCEF is summarized in Table 2. The safety review has well been advanced for the first commercial reprocessing plant (INFL) and these experimental data will additionally clarify safety margin in the design. The data on the TRU measurement in NUCEF will directly contribute to set up the classification of TRU waste in Japan.
Table 2  Expected contribution to safety assessment by the experiment in NUCEF

<table>
<thead>
<tr>
<th>STACY/TRACY</th>
<th>BECKY</th>
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<tbody>
<tr>
<td>1. Clarifying safety margin</td>
<td>1. Verifying process safety</td>
</tr>
<tr>
<td>- Criticality benchmark data</td>
<td>- Behaviors of iodine and minor actinides</td>
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<tr>
<td>- Confinement of radioactivity</td>
<td>- Confinement of radioactivity</td>
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<tr>
<td>2. Development of assessment methodology</td>
<td>- Verifying abnormal process behavior</td>
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<tr>
<td>- Scenarios of postulated criticality accident</td>
<td>2. Scenarios of TRU waste management</td>
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<td>and accidental excursion phenomena</td>
<td>- TRU measurement in solidified waste</td>
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<tr>
<td>- Assessment of complicated plant systems</td>
<td>- Leaching and migration behavior in</td>
</tr>
<tr>
<td>3. Development of sub-criticality measuring</td>
<td>promising disposal environment</td>
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<tr>
<td>technique</td>
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</table>

(3) Regulatory guides and its technical bases

Licensing review on nuclear fuel facilities including reprocessing plants in Japan is carried out principally based on the following regulatory guides:

i) Basic Regulatory Guides for Licensing of Nuclear Fuel Facilities (1980),
ii) Regulatory Guides for Licensing of Reprocessing Plants (1986),
iii) Target Doses for Plutonium In take in Relation to Siting Assessment of Nuclear Fuel Facilities (1983)

etc.

The regulatory guide for nuclear reactor are also applied to the licensing review on nuclear fuel facilities concerning other items including site, environment and prevention of disasters. Each regulatory guide has a detailed explanation for the application of each criteria. The technical standards for the design and methods of construction are shown by Order of Prime Minister’s Office. Handbook and data obtained in safety research and safety demonstration test are referred to the other technical fields needed for regulations of nuclear facilities. They contribute to the advancement and substantiability of technical base of the regulatory guides.

The first commercial reprocessing plant in Japan finished its licensing review which was carried out along with the above regulatory guides. Now activities to obtain the sanction of design and methods of construction on the reprocessing plant are being conducted. The NUCEF project will proceed safety researches, not only to ensure and demonstrate the safety of reprocessing plant concerning the sanction of design and methods of construction as well as its safety operation, but also to contribute to the substantiality of regulatory guides for future reprocessing plants.

Main subjects for the safety research in NUCEF, which are related to regulatory guides and its technical bases, are criticality safety, confinement of radioactivities and safety evaluation. Expected contribution of those researches are listed up in Table 3.
<table>
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</thead>
<tbody>
<tr>
<td>Criticality safety (Guides 10/11)</td>
<td>(1) Substantiality of guide interpretation of reasonable safety margin etc.)</td>
<td>Substantiality of Criticality Safety Handbook based on experimental data</td>
</tr>
<tr>
<td></td>
<td>(2) Logical provision of criticality alarm systems (Detection of sub-criticality etc.)</td>
<td>(Supplement of anticipated transients, reasonable safety margin and modification of analysis codes, increase of preciseness in data)</td>
</tr>
<tr>
<td>Confinement of radioactivities (Guides 2/4/7)</td>
<td>(1) Substantiality of guide interpretation for application of the principle of ALARA (Simulation, data base and so forth)</td>
<td>(1) Substantiality of standards for confinement of cells, glove boxes, hoods and their equivalents</td>
</tr>
<tr>
<td></td>
<td>(2) Substantiality of the safety assessment of dissolution of high burn-up fuels and MOX fuels (Basic data etc.)</td>
<td>(2) Substantiality of standards for monitoring to the measurement of low concentrated radioactive materials</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(3) Investigation of standards for confinement in facility (Provisions above mentioned are set by supply of basic data)</td>
</tr>
<tr>
<td>Safety evaluation (Guides 3/12)</td>
<td>(1) Rationalization of the evaluation methods for criticality accident (Analytical model, data base etc.)</td>
<td>(1) Estimation of the size of postulated criticality accidents (Methods for the calculation of degree and source terms)</td>
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<tr>
<td></td>
<td>(2) Substantiality of the safety assessment of confinement at anticipated transients, DBE and postulated accidents (Simulation, basic data etc.)</td>
<td>(2) Establishments of the confinement efficiency and evaluation methods for accidents</td>
</tr>
<tr>
<td>Storage of TRU waste (Guide 8)</td>
<td>Establishment of the guide for TRU waste management (Establishment of categorized values)</td>
<td>Establishments of standards for measuring techniques for TRU containing in radwaste</td>
</tr>
</tbody>
</table>
4. OUTLINE OF NUCEF FACILITIES

(1) Configuration of buildings

Figure 2 illustrates a bird’s eye view of NUCEF Building. There are three buildings in NUCEF: Administration Building, Experiment Building A and B. Each building has three stories and one basement. Experiment Building A occupies about 9,500 m² floors and has STACY (Static Criticality Experiment Facility), TRACY (Transient Criticality Experiment Facility) and a nuclear fuel treatment system for those critical facilities. In this building the criticality safety research is carried out. Experiment Building B, named BECKY (Back–end Fuel Cycle Key Elements Research Facility), is furnished with high density concrete shielded cells (alpha–gamma cell), many glove boxes and hoods. Study on the fundamental safety researches and advanced technology of fuel reprocessing, research on TRU waste management and studies on TRU chemistry are carried out in the building with an area about 8,000 m². Administration Building has an area about 3,800m² and laboratories for cold experiments as well as offices are located in the building 7,8. Figure 3 shows an aerial photograph of completed NUCEF Building.

(2) Facilities for criticality safety researches

STACY and TRACY are respectively installed in cells. Especially, STACY is contained in a hood within the cell to avoid spreading of plutonium contamination, which is based on the principle of multiple–containment. Table 4 lists up the specification of STACY and TRACY.

In STACY homogeneous and heterogeneous cores can be set up, alternatively. The homogeneous core uses both low–enriched uranium nitrate and plutonium nitrate solutions. The experiment is carried out using core tanks with the shape of either cylinder, slab, annulus or others. The heterogeneous core employs cylindrical tank with the combination of low–enriched uranium nitrate solution with low–enriched uranium fuel rods. The maximum thermal power is restricted to 200 W. The reactivity is controlled only by feed and drainage of the fuel solution. Neither control rod nor cooling system is equipped because of its easy fuel–solution transfer and low power operation.

TRACY has only homogeneous core consisting of low–enriched uranium nitrate solution. There are two operation modes for TRACY: the transient mode in which reactivity addition up to 3S is available and the steady state mode which regulates the excess reactivity less than 0.8S. The maximum thermal power is 5000 MW and 10 kW depending on the operation mode as shown in Table 5. There is no cooling system equipped in the facility. The reactivity is controlled by feed and drainage of the uranium nitrate solution and by a transient rod made of boron carbide. Two cylindrical cores are prepared for the experiments during the first phase of the experiment.

The fuel solution for STACY and TRACY is prepared in the fuel treatment system attached to the critical facilities, which consists of dissolution, concentration, mixing and purification units. The system adjusts chemical composition of all the fuel solution to be used. Each process unit is contained in glove boxes considering low radioactivity of the fuel solution. In addition, alpha–chemical laboratory and
in-line test loop are furnished for the analysis of the fuel solution and for the process control in the fuel treatment system. Criticality safety research on actinide separation process is carried out in the alpha-chemical laboratory with six glove boxes.

(3) Facilities for fuel cycle back-end research in BECKY

The alpha-gamma cells are installed in BECKY, together with twenty-one glove boxes containing the rigs for preliminary and supplemental experiments. Studies on reprocessing, TRU chemistry and TRU waste management are conducted with the facilities. Table 5 summarizes major specification of BECKY.

Alpha-gamma cell

The alpha-gamma cell consists of two cells, named process cell and chemical cell. Major part of study safety and advanced technology of fuel reprocessing are carried out in the process cell. And some of studies on TRU solution chemistry is carried out in the chemical cell.

The process cell is equipped with four pairs of manipulators and one in-cell power manipulator, and it contains bench scale experimental equipment for dissolving, off-gas treatment, extraction and liquid waste recycling process units. In this cell spent fuel specimens up to 45,000 MWD/T (Max. 3 kg/year) can be handled. The cell is also equipped with the apparatus for partitioning process. Actual high-level liquid waste (Max. 185 TBq/year) can be used in the process. The chemical cell is designed to accommodate various type of beaker-scale experiment rigs which are handled remotely using a pair of master-slave manipulators and an in-cell crane. Small amount of spent fuel up to 72,000 MWD/T can be treated in the cell. Besides those two cells, the loading cell is located next to the process cell, which is used for the receipt and storage of spent fuel specimens and for the management of highly contaminated solid waste arisen from the cells.

Laboratories for the research on TRU waste management

There are two rooms prepared for the research on TRU waste management. One room is furnished with a system for nondestructive measurement to be used in the studies on nondestructive and testing techniques of TRU waste. The system has its specialty in which a very high-detection sensitivity for neutron is expected by an improved electric circuit and by employing a lot of detection heads surrounding waste container with a practical size. The information obtained by the neutron detection is combined with data on absorption and refraction characteristics observed by a gamma-ray computed tomography installed in the room, so as to determine the contents of actinide in the solid waste with reliability. In the experiments simulated wastes made of solidifying materials and TRU test pieces are prepared. Another room accommodates nine glove boxes containing equipments, such as migration test rig and hot-press rig to produce new ceramic waste form containing TRU elements. Those rigs are used in the research on the treatment and disposal technology of TRU waste.
### Table 4 Major specification of STACY and TRACY

<table>
<thead>
<tr>
<th>Item</th>
<th>STACY</th>
<th>TRACY</th>
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<tr>
<td>Thermal power</td>
<td>200 W (Max.)</td>
<td>10 kW (for stationary operation)</td>
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<td></td>
<td></td>
<td>5000 MW (for transient operation)</td>
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<tr>
<td>Maximum fuel inventory</td>
<td>Uranyl nitrate solution 5 or 4 % enrichment</td>
<td>500 kgU plutonium</td>
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<tr>
<td></td>
<td>others</td>
<td>90 - 150 kgU</td>
</tr>
<tr>
<td></td>
<td>Plutonium nitrate solution plutonium</td>
<td>60 kgPu</td>
</tr>
<tr>
<td></td>
<td>UO₂ fuel rod 5 % enrichment</td>
<td>400 kgU</td>
</tr>
<tr>
<td>Nuclear limitations</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maximum excess reactivity</td>
<td>0.8 $</td>
<td>0.8 $ (for stationary operation)</td>
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<td></td>
<td></td>
<td>3 $ (for transient operation)</td>
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<tr>
<td>Reactivity control method</td>
<td>Feed and drainage of solution</td>
<td>Feed and drainage of solution, and</td>
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<td>(safety rods for shut down)</td>
<td>withdrawal of transient rod</td>
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<tr>
<td>Maximum reactor vessel</td>
<td>1.1 m³</td>
<td>0.5 m³</td>
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<td>volume</td>
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### Table 5 Major specification of BECKY

<table>
<thead>
<tr>
<th>Item</th>
<th>Research on advanced reprocessing process</th>
<th>Research on TRU waste management</th>
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<tbody>
<tr>
<td>Contents of research</td>
<td>- Improvement of PUREX process</td>
<td>- Development of TRU measuring technique</td>
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<tr>
<td></td>
<td>- Development of partitioning process</td>
<td>- Development of TRU waste treatment and disposal method</td>
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<td></td>
<td>- Related fundamental chemistry on TRU</td>
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<tr>
<td>Experiments method</td>
<td>Laboratory scale experiments using small quantities of spent fuel and high-level radioactive liquid waste.</td>
<td>Laboratory scale experiments using artificial and real TRU waste</td>
</tr>
<tr>
<td>Major radioactive materials and their Maximum handling quantities</td>
<td>- Spent fuel (about 45000 MWd/T) pellets and solution etc.</td>
<td>- Plutonium solid, solution etc. 10 g/year</td>
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<td></td>
<td>- High-level radioactive liquid waste 2 liters 185 TBq/year</td>
<td>- Radioactive isotopes (TRU, Cs, Sr etc.) solid, solution etc. 2.56 TBq/year</td>
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<td>Major equipment</td>
<td>- Alpha–gamma cells</td>
<td>- Glove boxes</td>
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<td>- Glove boxes</td>
<td>- Hoods</td>
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<td>- Hoods</td>
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</table>
Figure 2  Bird's eye view of NUCEF Building

Figure 3  Completed NUCEF building
5. CURRENT STATUS AND FUTURE PLANNING OF NUCEF PROJECT

(1) Current Status and Planning

Table 6 shows projected schedule of NUCEF project. Current Status and future planning of the facility is summarized below.

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<td>Hot Experiments</td>
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</table>

Construction

Manufacturing was started in 1988, from such equipments needed long time to be completed as annular tanks for solution fuel and glove boxes. Since then, various systems and equipments have been made and assembled. In 1991 installation of completed systems and equipments was initiated. In parallel with the installation work, pipe working and electrical wiring for the systems and equipments have also been conducted. At present, installation work was completed and has been followed by the cold function tests of all the systems and equipments toward the hot operation in the beginning of 1995 (calendar year) as shown in Table 6.

Licensing

NUCEF is divided into two kinds of facilities; the nuclear reactor facility and nuclear fuel facility, to content with the relevant regulations for licensing application. The nuclear reactor facility includes STACY, TRACY and the fuel treatment system. On the other hand, the nuclear fuel facility is BECKY. Licenses for construction and manufacturing of both facilities were obtained by October 1988 after government safety reviews. Coming next was the action for the construction permit for buildings,
various systems and equipments of the nuclear reactor facility. This was carried out by dividing into seven stages. In May 1991, all the permits related to uranium experiments were obtained. Also, government inspection of the building, systems and equipments of facilities, which was initiated in 1989, have been carried out towards the permission of test operation at the middle of 1994 in such a way that can meet the progress in the building construction and manufacturing of systems and equipments in off-site factories.

Safeguards Arrangement

It is necessary that a relevant safeguards arrangement is agreed upon with IAEA through STA and a consent for this agreement is given by United States for the utilization of nuclear fuels in NUCEF. Discussion with IAEA and the Nuclear Safety Bureau of the STA started from 1988, and after DIQ (Design Information Questionnaire) was submitted in 1992, FA (Facility Attachment) was agreed in 1993, respectively. The DIV (Design Information Verification) is continued by the IAEA.

(2) Research Collaboration

Researches in NUCEF will cover wide range of research field in the back-end of fuel cycle. It becomes very important to carry out the researches, not only under the domestic cooperation with the government, university and industries, but also under international cooperation as shown in Figure 4. By the cooperation system which is summarized in the Planning Committee of NUCEF, preparation for the concrete cooperation has been proceeded, and some of them have just started. In order to make the research program transparent and effective, the first NUCEF Seminar (domestic) and the first International Symposium will be held at NUCEF, March and October 1995, respectively. These first Meetings will subject to Engineering Safety for fuel cycle safety; criticality, confinement of radioactivities and postulated accident. Through such cooperation, many researches will effectively be conducted at the facility and it is expected that NUCEF will become one of the centers of safety research of fuel cycle back-end.
6. Conclusions

For the safety research for the plant and the fundamental research upon advanced technology for the fuel cycle back-end facilities, new experiment facility NUCEF has just completed in its construction and will be commissioned the hot operation at the beginning of 1995 (calendar year).

In the facility, criticality safety research with STACY and TRACY, safety research and research for advanced technology on reprocessing including TRU partitioning from HLW and safety research on TRU waste management in BECKY are planned. By those researches, contributions to the enhancement of technical bases for the safety guides and improvement to safety assessments of nuclear fuel cycle back-end facilities are expected to be made through criticality safety research, research for confinement of radioactivities, research for postulated criticality accidents or other accidents and research on TRU waste measurement technology.
For the effective use of the NUCEF and for the further acceleration of research activities in nuclear fuel cycle back-end field, research collaboration are widely planned among domestic and international phases. On October 1995, international symposium is to be held at NUCEF focusing to the field of Engineering Safety on nuclear fuel cycle back-end.

By these cooperation researches, technical bases of fuel recycling, especially its safety will be enriched and contribution will be also expected to steadily promote the fuel recycling.

Acknowledgment

The authors wish to present sincere appreciation to the participated members of NUCEF Project for the facility construction and discussion of research program. Thanks are also due to Mr. Maeda, A., Dr. Mineo, H. and Mr. Yanagisawa, H. for the support in the preparation of the present paper.

References

SUMMARY : ABSTRACT

The ‘modus operandi’ of the Fast Reactor Reprocessing Plant (FRRP) is briefly discussed together with how that has developed against a programme of varying objectives and standards. Attention is given to the original design principles and how performance has been affected by these; developments over the 15 years are then identified. Particular attention is given to nuclear materials safeguards, both system and performance. The licensing (1990) risk assessment is discussed, in particular in comparison to the original 1972 - 1978 hazard assessments. Later more recent events are also discussed.
1. **INTRODUCTION**

1. The fast reactor reprocessing plant was constructed in the late 1950s and operated into the early 1970s with continued process and safety developments. In the 1970s the plant was extensively decommissioned and reconstructed to extend its capability to the reprocessing of irradiated MOX fuel from the Prototype Fast Reactor (PFR). It has continued in that role since 1980.

2. PFR reprocessing commenced in 1980 and since that time both the process and plant equipment have been modified in some areas to accommodate technological and safety improvements and to meet varied process research and development requirements. Additionally, unirradiated fuel manufacturing residue material has been recovered using the plant, and more recently a ‘head-end’ extension.

**The Reprocessing Plant**

3. The reprocessing plant is located in the Fuel Cycle Area (FCA) of the Dounreay site and occupies two buildings situated on either side of a central access corridor. The layouts are shown in Figures 1 and 2.

2. **DESIGN PHILOSOPHY**

4. The design of the reconstructed plant was led by the operational and production requirements. However, the approach was constrained by the need to meet greater stringency of the radiological protection standards where, for instance, individual personnel dose exposures were (then) required to be no more than 1-2 Rem/year.

**Basic Requirements**

5. In addition to accommodating the increased plant throughput (from 1 to 3½ tonnes/year HIM) and meeting the flowsheet requirements, the prime considerations dictating the scope of the modifications and design were identified as:-

(i) improved containment, both primary vessel and cell; (ii) rearrangement of the solvent extraction equipment to accommodate the two macrosolute flowsheet; (iii) provision of new cave facilities for the far more complex PFR fuel assembly breakdown; (iv) provision of a new dissolver and feed liquor clarification equipment; (v) increased ‘head-end’ radiation shielding; (vi) increased liquor tankage to improve fissile material accountancy for both fuel cycle evaluation and to meet ‘safeguards’ requirements; (vii) improved ventilation arrangements; (viii) improved waste transfer and handling systems; (ix) improved liquor sampling and sample transfer arrangements; (x) the reduction of in-cell maintenance requirements; (xi) the incorporation of more and improved on-line instrumentation; (xii) the incorporation of detailed operational improvements identified as being desirable over 15 years of DFR operation.

335 Ref: CADPAP1
Design Features

6. The plant was required to deal with 180 day cooled fuel at 3 kW heat rating, but to recognise a potential for operation with 120 day cooled fuel. The fission product decontaminated plutonium was to be essentially uranium free. The depleted uranium was to be recovered but not immediately returned to the fuel cycle.

7. The fuel breakdown technique was based on the utilisation of a geometrically limited batch dissolver, and the minimising of the generation of plutonium contaminated waste materials (PCM). This led to the concept of making access to single pins and then their cropping into the leach dissolver system. Laser cutting of the sub-assemblies wrapper was adopted in view of the inherent in-cave simplicity and capability, and the avoidance of generating active swarf and powder from mechanical sectioning. The projected presence of fission product insolubles as noble metal based alloys led to the inclusion of a feed liquor clarification stage using high speed centrifuges.

8. Special consideration was given to input and export fissile material accountancy as part of an overall critical review of accountancy across the whole plant. This was required to satisfy stringent internal fissile material accountability and the increasing stringent demands of the international safeguards regime, and also to assess overall plant process efficiency. The disassembly cave was also to contain neutron-interrogation equipment for the assay of all solid waste for fissile material content.

9. The solvent extraction section made extensive use of existing small geometrical limited mixer-settlers. Therefore, the new designs centered on peripheral plant items. The containment philosophy was to locate the plant within cells with all pipes penetrating into process vessels being terminated out-of-cell. Remote instrument devices located in sealed thimbles, were used extensively.

10. Pumping systems using fluidic devices were developed and installed for some of the liquor movement, particularly on recycling solvent; other systems include gravity, steam ejection and air lifting. A semi-automatic remote sampling system was installed and incorporated pneumatic transfer arrangements to link the plant and laboratory without crossing containments.

11. Taking advantage of Vortex Amplifiers (VXA's) devices, a low flow, controlled depression filtered active ventilation system was installed. The incentive for reducing active air flows came mainly from the problems of changing, storing and disposing of filters coupled with past experience of high flows and velocities distributing activity. The vortex amplifier acts as a "no moving parts" control valve with a feed loop capable of maintaining a controlled depression in the system over a wide range of flows.

12. The waste handling procedures were dramatically improved by the utilisation of engineered mechanical 'double-door' systems, superseding the PVC bag welding facilities.
Operational Features

13. An aspect demonstrated in the first PFR run was that the employment of personal ‘dose-rate’ audible alarm units (‘bleepers’) continued to contribute to dose reduction and the ALARP principle. The value of such equipment was demonstrated in the former DFR reprocessing mode and more importantly during the decommissioning. This was extended to include the wearing of Personal Air Sample (PAS) units; the alarm function being covered by installed static sample/alarm units throughout the plant.

14. A significant operational safety feature, demonstrated as early on the first PFR campaign, was the extension of the control/ access zone right up to the cell and cave faces. This gives unfettered access to the operators and more importantly to the supervisors, surveyors, and managers such that ‘housekeeping’ is improved.

Design Safety

15. The approach to the design of the reconstructed plant was essentially two fold: to ensure that the additional feature required for technical reasons were meeting modern - and projected - safety standards, and to accommodate design features that the earlier studies had indicated required attention but where access constraints had precluded them being tackled. Design requirement and their intended resolution were required to be submitted earlier to a Safety Working Party (SWP) - a committee of operational, safety and engineering specialists.

16. Safety was addressed first by the establishment of a suite of general Design Safety Principles (DSPs) produced by the AEAs internal safety department in consultation with the plant operational management team. These were discussed, amended and agreed, to become the design safety ‘law’. These general DSPs were ‘top tier’ statements that were developed into more detailed DSPs for each section of plant as the design was developed.

17. The approach to the formulation of safety principles during design, the formal assessment of the design, and the safety assessment of operations, were embodied in a documented approach to risk assessment. This, in addition to giving guidance on the identification and evaluation of hazards, the identification of standards and the formulation of “Design Safety Principles”, additionally introduced the use of Probability Consequence curves as a basis for reliability analysis and the acceptability of “Risk”.

18. The aim of compliance with the Design Safety Principles (DSPs) was to demonstrate that safety had been adequately considered in the design process. It was recognised that the design of any plant should not place undue reliance on operational safeguards and managerial control; engineered safety features should be used as a first line of defence, together with the exploitation of intrinsic safety features or any transient limit to inventories. Overall approximately 750 DSPs were generated covering all of the 15 separate sections of the plant.

19. On the completion of the design, the design office were required to generate a Design Safety Report (DSR) identifying how the design complied with the DSP requirements. That document was subject to review by the operators as it was developed (as was the design itself).
Accountancy and Safeguards

20. When it was decided to reprocess fuel from the PFR in the plant which had reprocessed enriched uranium from the DFR reactor, there were fundamental differences which had to be addressed. The largely soluble metallic uranium alloy fuel would be replaced by a plutonium/uranium mixed oxide fuel where it was known that there would be measurable arisings of plutonium in the insoluble material resulting from the dissolution stage. The mechanical breakdown of the PFR fuel sub-assembly was far more complicated than the fuel rods from DFR.

21. International safeguards had not existed when the plant was originally designed and built, and no inspection from any safeguards regime had occurred from 1959 until 1973, when Euratom started inspection activities at Dounreay. The input verification on DFR fuel had been satisfactory from a domestic accountancy standard, but the more sensitive plutonium fuel meant that a new input accountancy measurement point was required. Measurable weights of plutonium in the solid waste arisings from the mechanical breakdown of the fuel and its subsequent dissolution also meant that improved NDA methods for measuring the plutonium in that waste were required.

22. Accountancy of uranium and plutonium in the aqueous waste streams are identical in approach to those of the solvent extraction process for enriched uranium, and consist of a bulk volume measurement in the plant in conjunction with the analytical determination of uranium and plutonium concentrations in discrete samples.

23. In the solvent extraction process, the uranium and plutonium are separated, and the plutonium product transferred to a concentration facility. Here, the concentration of the plutonium is increased from 25g/litre to 300g/litre by evaporation.

24. The concentrated plutonium nitrate is exported from the site to the fuel fabrication source, and the accountancy measurement is a bulk weight in conjunction with the concentration of plutonium in samples taken during the filling of the export vessel.

25. Plutonium in solid waste arisings from the fuel dissolution are now measured by an active neutron technique employing a $^{252}\text{Cf}$ shuffler. Sub-assembly wrappers and other solid waste where the plutonium content is negligible are assayed using passive neutron counting techniques. Solid waste arisings in the chemical separation and plutonium concentration stages of the process are assayed for plutonium content, again employing passive neutron counting.

26. Wherever cooling water is used in the plant, gamma monitoring at the plant exit points ensure that plutonium does not leave the facility unmeasured. Dissolver scrubber liquors are sampled for radioactivity and plutonium analysis.

27. The plant is divided into two Material Balance Areas (MBA) for International Safeguards. The reactor advised weights of uranium and plutonium are the main input to the first MBA (QDR8), and the measured output from the input accountancy measurement point is the main input to the second MBA (QDR9).
28. In addition to the solid and liquid waste arisings, other issues include fuel pins and specimens to PIE laboratories, and samples for accountancy and plant control purposes to the analytical laboratories. Fuel pins from PIE and some returns from the analytical laboratories are similarly accounted for as receipts to the appropriate MBA.

29. The input accountancy tank in QDR8 is a ‘b’ shaped vessel with an accurately calibrated top limb and bottom sump extension where the volume measurement is known to be accurate to 0.1 litre. Transfers to QDR9 are arranged so that the initial volume is always in the top limb and the final volume is measured in an extension of the bottom limb where the associated error in the volume measurement is also negligible. A typical full tank volume is 170 litres and final volume, after transfer, is typically 0.5 litres. The dissolved fuel after centrifuging is transferred through the accountancy tank in batch mode to the solvent extraction buffer feed tank.

30. Twice a year a Physical Inventory is recorded and a Material Balance Report derived in the conventional manner. The Physical Inventory is measured after the plutonium content of the dissolver and other head end vessels has been reduced to negligible concentration by successive washes of nitric acid, and the solvent extraction part of the plant has been similarly washed out. Finally the plutonium evaporator is emptied and the inventory of plutonium nitrate is measured in calibrated tanks in QDR9.

31. As far as practicable all waste in the fuel disassembly area is collected and the plutonium content measured by active neutron interrogation. Finally, the entire plant is inspected by trained operators to ensure that every location known possibly to contain nuclear material has been sampled and the nuclear material inventory measured. Equally importantly, the plant is examined to ensure that no unrecorded inventories of nuclear material are present.

Safeguards

32. Euratom safeguards was already well established on the Dounreay site before the commencement of PFR fuel reprocessing. From 1973 to 1979 they had inspected eight reprocessing runs on DFR fuel amounting to 1.5 te of HEU.

33. Before the new cave and cells were actively commissioned with PFR fuel they, and the pipework, were subject to close scrutiny by Euratom and IAEA inspectors. The purpose of the inspection was to verify that the new parts of the plant installed for PFR fuel reprocessing were as described in the Basic Technical Characteristics, supplied in advance to the Safeguards Inspectorate. A visual examination of the process layout in the solvent extraction section provided evidence that no diversionary routes out of the main process line were present.

34. The Safeguards philosophy applied to the plant by Euratom, consists of sealing all major external entry and exit points from the process when reprocessing was in abeyance. As the exit points from the solvent extraction plant are numerous and difficult to seal, the drive shaft to the Constant Volume Feeders are sealed, making it impossible to operate the solvent extraction process.
35. When reprocessing starts, the seal on the fuel entry position is removed together with the seals on the solvent extraction process. Paper seals are placed on the flask containing the sub-assembly for the short journey from the reactor. The number of pins in each sub-assembly is verified by the inspector and cross-checked with the video recording taken by the operators. Euratom inspectors witness batch transfers from the input accountancy tank to the plant feed tank to ensure no diversion of material in the plant head end. The analysis of these samples is carried out by the VOPAN technique which is also witnessed by Euratom.

36. The plutonium product is subject to strict inspection by Euratom; the weight of the empty flask, and then after filling, are recorded and the laboratory analysis is witnessed and reference samples taken as required. The nuclear material content of any sample point in the plant may be subjected to the same attention from the inspectors, and the activity extends to the NDA measurement of the solid waste arisings. Every month the measurement of nuclear material in the low level plutonium waste treatment MBA is inspected, as a further verification on the declared uranium and plutonium sent to the Waste Areas MBA for further treatment prior to transfer to retained waste or discard.

3. PERFORMANCE

Operational

37. The active operation commenced in early September 1980 and the 1 ½ tonne (Heavy metal) campaign was successfully completed in December. It was fully reported in the autumn of 1981.

38. Up to the end of 1994 the plant had conducted a total of nineteen MOX campaigns including three addressing unirradiated residues. Over this twelve year period some 23 te of uranium and plutonium has been processed through the plant and over 3.5 te of plutonium have been exported to Sellafield. The essential details appear in Table 1.

39. Fuel, up to 17.6% burn-up, and cooling, down to 140 days after discharge from the reactor, has been processed with little operational difficulty. The plant has demonstrated a high plutonium recovery from the irradiated fuel with over 99.5% of the plutonium being returned for re-fabrication into fresh fuel for the reactor. The first recycled plutonium was loaded back into PFR in 1982. The liquid and solid waste streams from the plant have contained very low amounts of plutonium and the total amount of radioactivity discharged to sea has averaged less than 10% of the permitted discharge. The introduction of the standardisation of solid waste packaging systems, using engineered routes and reusable containers, and NDA plutonium measurement methods have reduced secondary waste generation and provided the most effective use of the site retrievable stores.

40. Consequent upon the UK setting aside Fast Reactor development with the consequential closure of PFR on the 31st March 1994 the operational objectives for the Fast Reactor reprocessing plant have been modified. In addition to reprocessing the final core fuel, it was decided to reprocess the plutonium dilute (only containing inbred material) breeder fuel from both the axial blanket components of the fuel sub-assemblies, and the radial blanket.
assemblies. This process has now been undertaken without difficulty once the criticality safety consideration associated with the reduced higher Pu isotopic levels (neutron poisons) had been addressed.

41. Arising from the former plant decommissioning exercise, it was possibly to identify design features which would promote later plant decommissioning activities. Some of these were introduced into the reconstructed plant and have already been beneficial. Instances include the modular concrete shielding on the dissolver sub-cell, the installation of stainless steel cat-ladders in the cells, the provision of separate shielded cave crane maintenance facilities, and the provision of special drainage to permit high pressure water jetting decontamination.

42. The plant has progressively improved its safety performance over the three decades of its fully active operation. This is judged against all relevant factors - routine dose to the operators, dose to the public from aerial and liquid discharges, and the incident and accident frequency within the plant. The prime developments over the early years of the plant were targeted at resolving technical difficulties and meeting additional requirements. The safety performance is not discussed further here as it is outlined in the companion paper at this topical meeting.

4. **SIGNIFICANT DESIGN DEVELOPMENTS**

**Fuel Disassembly**

43. The Fuel Disassembly Cave is equipped with five zinc bromide filled windows and pairs of master slave manipulators with a number of basic machines. The basic machines consist of a fuel receipt port, sub-assembly handling machine, universal handling machine, pin shear machine and laser cutting machine.

44. Initially, a 400 watt laser was installed to cut open the transit can and the wrapper to assist in gripping the pin ends. A prime requirement of this process is to be able to cut the wrapper without damage to the adjacent pins. This process has been very successful. The original 400 watt laser was working close to its limit, and hence there were some reliability problems. In 1988 a 1.2 Kw laser was installed, with the initial in-cave light guides being retained. This has proved to be a very robust and reliable device. It is also used for sectioning the sub-assembly wrapper to assist in waste disposal.

45. Fuel pin cropping for the fast reactor pins has been focused on double pin shearing, although other designs have been tried. The installed Hezel cropping machine was a hydraulic fluid powered machine which operated well until the in-cell hydraulic cylinder failed. Although the cylinder was successfully replaced the ingress of contamination into the hydraulic lines made continued use of this machine impossible due to high radiation levels. This unit was replaced by an air driven rotary cropper which, although very effective, with easily replaced modules, does result in contaminated air entering the cell and leads to a fine coating of dust and oil on the adjacent cave surfaces and windows. Initial attempts to limit cell contamination through the use of cropper dust covers and directed airflows with dust
collection systems were ineffective and abandoned. Future cave design should provide for cropping in a separate area to sub-assembly dismantling to limit contamination and ease waste measurement problems.

46. An important aspect of fuel disassembly operations is the handling, classification and disposal of the waste steels from dismantled sub-assemblies. The cave contains a “Californium Shuffler” device, which replaced the previous 14 MeV neutron generator to operate a neutron interrogator for the measurement of fissile material in solid waste streams. The $^{252}$Cf source has now been replaced in this device and, at the same time, the design of the array neutron detector tubes has been improved so that tube replacement is now effected from outside the cave. The system continues in a totally automated mode.

47. A passive neutron counting system has been installed in an adjacent cave to the PFR fuel disassembly cave. This system measures the spontaneous fission and alpha-n neutrons from residual Pu contaminating the waste steels. The detection limit can be reduced by extending the counting period. This allows waste steels, and other cave wastes, to be classified as ‘low alpha”; the alpha level corresponding to the UK alpha limit on low level waste (LLW) of 4 GBq/tonne.

48. This neutron counting system is backed up by a ‘slab’ counter. This is a most useful device which uses the flask gamma shielding to prevent interference of the neutron detectors from high energy gamma emitters from $^{60}$Co present in some sub-assembly components. Once again by extending the counting time the limit of detection can be reduced to below the UK LLW alpha limit.

Fuel Dissolution and Clarification

49. The fuel dissolver has continued to function very satisfactorily. It is subject to routine annual internal inspection by a remote CCTV system. End grain attack on dip pipes has been experienced but dissolver body corrosion is minimal, indicating a life of at least a further ten years at current throughput rates.

50. The centrifuge based dissolver liquor clarification system continues to function well. The centrifuge has an automatic speed controller which has simplified its operations. Although amounts of insolubles collected in the centrifuge bowls are lower than anticipated it is still practice to change the bowl after every dissolution.

51. Dissolution rates have been followed by accurate SG measurements of the dissolver liquor. These indicate that typically 99% of the fuel is dissolved within 2-4 hours reflux and dissolution is complete after 8 hours. Analysis of many hundreds of dissolution results have indicated the following:-

- Pu insolubility has a mean value of 0.7%
- there is no correlation between irradiated fuel solubility and the fuel manufacturing route
- there is no evidence of increased insolubility with burn up
- there is no evidence of increased insolubility as the Pu to U ratio increases
- pre-irradiated solubility tests give no guide to irradiated fuel solubility

Ref: CADPAPI
Following the operational experience gained with this dissolver, an additional unit has been designed and installed to deal with some of the more intractable unirradiated residues from the fuel fabrication route. This new dissolver suite is only lightly shielded and is based on the design development for the once proposed EDRP facility. The main dissolver feature is the access to the lowest sump point down the charge limb.

On-Line Instrumentation

The use of on-line instrumentation has been an objective of fuel reprocessing since the beginning of research and development in this field of work. The techniques which have been tried and developed are numerous. Many have evolved into devices making up part of today's plant while others have not realised the benefits expected and are no longer a part of the process monitoring equipment.

Ultrasonics

Several acoustic/ultrasonic measuring systems have been tried and tested in the reprocessing plant; the most notable are as follows:

- **Scrubber Liquor Flowmeter**

An ultrasonic flowmeter was installed in the dissolver off gas scrubber liquor pipework in 1984. The system consists of a pipework 'dog-leg' with an ultrasonic transducer at each end. A system has now been engineered to allow an alternative flowmeter (magneto inductive) to be inserted in the system should the existing one fail completely due to the age of its electronics.

- **High Activity Level and Concentration Monitor**

This system consisted of probes in each of the high active mixer-settler boxes. Each probe consisted of 4 ultrasonic transducers and a platinum resistance thermometer in a 'thimble' immersed into the liquor in the mixer-settlers. The system operates by timing and analysing ultrasonic pulses across a known distance in the solvent and in the aqueous phases and via a 45° reflector to the solvent surface and the solvent/aqueous interface. However, the system was never able to cope with the continuously altering levels, the air bubbles, the emulsion layers and sludges that are encountered in the plant mixer-settlers. They were replaced by more conventional equipment in the mid 1980s.

- **Heavy Metal Monitor**

This system was very similar. It operated on the same principles and had similar specially designed electronics. The system consisted of three parts, each measuring the speed of ultrasound across a \( \frac{1}{2} \)° nominal bore pipe using a 45° deflection mirror. The transducers
were fitted to the loaded solvent product lines and the unloaded solvent feed line, which was used as a reference. Again, the system never gave consistently-reliable results. It was unable to cope with the effects of temperature changes flexing the sample pipe, the pipe not being full enough at times, the very small air bubbles in the product and the effects of the plant sample systems.

Plutonium Breakthrough Monitors

58. The Plutonium Breakthrough Monitors (PBM) s are fitted adjacent to mixer-settler boxes in the Medium and Low Active Cells. They detect plutonium concentration and by their location warn the operators if the plutonium profile front drifts towards the raffinate end of the mixer-settler. The PBM s measure neutrons, each monitor consists of a group of three BF₃ proportional counters moderated and shielded from gamma radiation. During the early to mid 1980s there were several changes to the PBM. The number of monitors was increased from five to ten and the BF₃s were covered with cadmium sheet to give shielding from neutrons arising in surrounding mixer-settler sections. The system is still in use and plays a major part in the control of the solvent extraction process.

Inactive Feed Control System

59. Precise metering and control of liquid chemical flow is necessary to achieve the required separation efficiency. A manually set displacement pumping system in conjunction with a thermal flowmeter control device was initially utilised within the inactive feed system, but unreliability required that the system be upgraded. A system employing Constant Volume Feed (CVF) units was installed to replace the pumping system. The CVF units were designed with stepper motors driving rotating hollow shafts with radially connected open ended, pick-up tubes. Rotation of the CVF radial tubes allowed liquor collection from a trough and gravity transfer to the appropriate plant feed pipe. The rate of liquid flow could be determined from the angular rotation rate of the CVF and it was therefore not necessary to measure liquid delivery with a flow measuring device.

60. The feed control system was further improved to allow central control of the CVFs, provide mimic indication, and monitor the status of the system. The reprocessing plant data acquisition system was also incorporated into the modified arrangements to enable accurate volume balancing, across the extraction process.

High Active Cell Liquid Level Probes

61. The high active mixer-settler box ultrasonic probes were found not to be effective, so a conductivity thermistor invasive probe system was reinstated, albeit of a more modern automated design from the former units.

Oscillometric Density Meters

62. Anton Paar oscillometric density meters are widely used at Dounreay. The flowing plant stream remains contained within a stainless steel U-tube at all times in these instruments—an important safety advantage. The steel U-tube is clamped to a heavy solid support and the
tube is forced to oscillate by a piezo-electric crystal transducer. The tube oscillates at a characteristic frequency which depends on the mass of the tube plus its contents enabling the liquor density to be calculated. The main applications are for the determination of the strength of concentrated inactive feeds. However, such a meter forms part of the on-line accountancy system in the Residue Recovery Plant.

**K-Edge X-Ray Densitometry**

63. The absorption of X-rays by different elements varies mainly in a continuous and smooth manner as a function of X-ray energy. At certain energies, however, the X-ray absorption by an element increases in a step-like manner. The energy corresponding to a step change is equivalent to that required to eject a ‘core’ electron. If L-shell electrons are ejected the step change is called L-edge and for K-shell electrons the K-edge. The technique of K-edge X-ray densitometry irradiates a sample with ‘white’ X-rays from an X-ray generator and the transmitted rays are detected. The output from the detector shows the step-reduction in transmission due to uranium and plutonium in the sample. The K-edges for different elements have different energies. The heights of the K-edge are related to the concentration of the elements in the sample. X-rays corresponding to K-edge energies are used since they can penetrate steel without too much attenuation. It is therefore possible to design an on-plant measurement system without breaking liquor containment. Because of the high output from the X-ray source tube, the presence of gamma active species in the sample is not generally a problem.

64. K-edge X-ray absorptiometry is now an established technique worldwide for in-plant measurement of uranium and plutonium. It has the advantage of being non-invasive and non-destructive. It can be used as a discrete sampler analyser, a plant control analyser or, commonly, as an accountancy measurement. High precision is a characteristic of the technique. A K-edge X-ray densitometer forms part of the instrumental accountancy system in the new Residue Recovery Facility.

**Collimated Gamma Scanners**

65. The Collimated Gamma Scanners (CGS) are required to provide direct ‘real time’ measurement of the degree of fission product separation in liquor streams in Cycle I. The system provides an early warning in the event of changes in the solvent extraction process. A high resolution intrinsic germanium detector is positioned to detect collimated gamma rays from the loaded solvent being transferred from one of the High Active Cells. Four digital group selectors are used to calculate peak areas for the 4 fission products $^{106}$Ru, $^{137}$Cs, $^{95}$Nb, $^{92}$Zr.

**Power Fluidics**

**Liquor Transfer Systems**

66. Fluidic pumps are used inside shielded areas where maintenance free equipment is required. Fluidic pumps are in use on the dissolver off-gas recirculating scrubber, the dissolver scrubber liquor discharge system and on the recycle of Cycle II and III solvents.
pumps, having no moving parts, have proved extremely reliable in operation, being in use 24 hours/day for a duration of several months.

Ventilation Control

67. The ventilation of active caves, cells and gloveboxes is exhausted through primary and secondary filters before exiting the building to the stack. Booster fans and vortex amplifiers control the air flows and depression in these active areas.

68. The vortex amplifier is a three part device consisting essentially of a short cylindrical chamber with radial supply flow inlets, tangential control flow inlets, and an axial outlet. The particular characteristic of the device relevant to active ventilation is that depression is held substantially constant for a wide range of flows, and moreover they provide reduced resistance at zero depression, enhancing inlet flow through containment breeders.

Solvent Extraction

69. The solvent extraction flowsheet was designed to take advantage of the existing DFR reprocessing plant which was based on safe-by-geometry mixer-settlers. A three cycle flowsheet with final purification of the plutonium nitrate product in Cycle III was adopted. The flowsheet was unusual in one respect, in that a partial separation of uranium from plutonium was made in Cycle I, with approximately half of the dissolved uranium being discharged at this stage; this partial separation was carried out because it was believed that the existing Cycle II mixer-settlers could not handle the full heavy metal throughput without flooding. The plutonium was separated from the remaining uranium in Cycle II. The partition agent used in both cycles was sulphuric acid and the final plutonium purification cycle, Cycle III, removed the sulphate ion to give a pure plutonium nitrate aqueous product. There was no requirement to produce pure uranyl nitrate product; the uranyl nitrate was all transferred to a waste storage facility and it was unimportant that part of the uranium product had only received one cycle of fission product decontamination.

70. After a number of campaigns the operational solvent extraction flowsheet was changed in 1988 to carry all the heavy metal content forward from Cycle I to Cycle II and perform the complete partition of plutonium from uranium in Cycle II. The flowsheet still used a sulphate-plutonium backwash with a small addition of ferrous sulphamate. The uranium backwash feed was originally dilute sulphuric acid in order to limit flows through the plant mixer-settler, but this was later changed to dilute nitric acid to allow direct transfer of uranyl nitrate to other plants.

Safeguards Development

71. Euratom has used the Fast Reactor Reprocessing Plant to develop different safeguard techniques. They have introduced different types of seal in order to evaluate them in an operational facility. A Vacoss type seal was used to record the transfers to solid waste from the reprocessing plant flask exit point. Other seals have included acoustic types and stainless steel instead of the copper/brass in general use.
72. In 1992 the Residue Recovery Facility was commissioned with plutonium/uranium full fabrication residues and their purpose built additional dissolver head end of QDR8, which incorporated several new techniques for nuclear material measurement. These were funded directly by the DTI.

73. Conventional volume estimation of the accountancy tanks in this new plant was corroborated by a weigh bridge which it is hoped will eventually replace the volume measurement. On-line measurement of plutonium and uranium is made with a K-edge absorptiometer and an Anton Paar densitometer is also employed on-line.

Joint Euratom - IAEA Inspectors

74. The United Kingdom Government offered advanced technology nuclear facilities for inspection by the IAEA, and the Fast Reactor and Reprocessing Plant at Dounreay were designated by the IAEA. Joint inspection with Euratom started in 1980 and continued until 1982 when the IAEA announced that the outcome of their inspection had been considered satisfactory and the facilities were de-designated, which left Euratom as the sole International Safeguards presence at Dounreay.

Safety and Risk Assessments

75. The safety performance of the plant has generally confirmed the relevance of the mid 1970s design and the subsequent close managerial surveillance of the plant.

76. In the period 1988 - 1990 the safety performance was subject to a major detailed review due to changing UK regulatory requirements. It was necessary as part of that to produce more detailed Safety Cases that identified hazards, explained their mitigation and quantified residual risk. These risk assessments were carried out against standards adopted by the UKAEA and accepted by the regulators; the criteria include the limits that accidents from operation of a facility should not lead to premature fatality risks in excess of $1 \times 10^{-5}$ for plant operators and of $1 \times 10^{-4}$ for public external to the site. For radiological risks these can be translated to dose levels by virtue of the stochastic exposure: harm relationships - as identified by the ICRP.

77. The UKAEA established a format for such a Safety Case which was accepted by the NII for the process of licensing in 1990. It included requirements to address the management of safety, to review the plant safety performance, to identify all safety related equipment and mechanisms, to discuss the development of the plant and process, to discuss the facility in terms of modern standards and to generate a quantified risk assessment - preferably invoking probabilistic arguments. These predate the NII SAPs as formally issued two years ago, but lead to similar safety standards.

78. A systematic approach is required to be applied to the execution of the hazard assessment and it is required to be structured such that it can readily be examined and audited. An early stage of any hazard assessment is the application of a systematic technique for the identification of plant hazards and potential accidents. It is then necessary to identify the
possible causes and consequences of those hazards that could lead to a risk, and to quantify that risk.

79. A technique often applied at the design stage of a new plant is a Hazard and Operability (HAZOP) study. However, in 1988 HAZOP studies were not seen as the most effective way of either demonstrating or of promoting enhanced safety, particularly for plants having an operating history. The validity of detailed HAZOP studies might also be questioned in terms of plant that is largely inaccessible (behind concrete shielding etc). Consequently, the thirty non-reactor operating plants at Dounreay, including the reprocessing plant, were assessed by an alternative technique that conformed to a structured and systematic approach; the HARMs procedure was developed to meet this need and was used for the FR reprocessing plant risk assessment.

80. The underlying principle of the approach is qualitatively to rank the identified hazards in order of severity so that detailed quantitative risk assessments can then be carried out for those hazards which present the greatest risk. The risk posed by a particular hazard is quantified by firstly predicting the hazard frequency using fault tree analyses, which start with unwanted hazardous consequences and systematically identifies those combinations of events that could lead to its occurrence. The radiological consequence to the critical groups is then determined.

81. The HARM (Hazard Assessment Review Methodology) procedure first addresses previously identified hazards, (of radiological significance) such as those discussed in the Design Safety Principle (DSP) reports and any relevant definitively approved previous hazard assessments. It reviews, extends and updates them in light of any plant and process modifications and operational experience. Earlier safety principles are checked to assure that they have been carried forward into the design and also into the operating disciplines.

82. Following on the study of the documentation and procedures as discussed above, the consequences of plant malfunction that could produce an unacceptable radiological risk are identified. (Note ... in most cases the plant safety documentation addresses all hazards - not just those of radiological significance. The licensing is directed to radioactive risk). The faults that could lead to consequences are subject to an initial identification and then to a quantitative risk assessment; this is very much a “top down” approach. It has been found that the procedure has an extensive safety audit.

83. The HARMs approach was applied most successfully to the Fast Reactor reprocessing plant. The technique filters the postulated accident scenarios into High, Intermediate or Low fractions. It does that by identifying this with a matrix, relating High, Intermediate, Low and Remote frequencies with High, Intermediate, Low and Trivial consequences. The quantified risk assessment was first confined - as a matter of prioritisation - to the perceived High risks and then extended to include the perceived Medium risks.

84. Pursuing this assessment was primarily targeted at satisfying the regulatory bodies as to the safety of the plant and its operations. Some positive features emerged from the studies where it was seen that risks could be reduced by relatively minor engineering attention and by detailed changes to some of the operational practices.
85. In more detail the individual risk of premature fatality to a member of the workforce from the assessed internal fault conditions is $1.5 \times 10^{-6}$ per year for hazards categorised as potential high risk and $1.5 \times 10^{-7}$ per year for hazards categorised as intermediate risk. The high risk hazards are dominated by excessive doses received from manipulator decontamination, accidental detachment of gloves from gloveboxes, and Pu leakages leading to a criticality in the product plant. These risks are currently being reassessed as they are considered pessimistic. The factor of ten difference in the total predicted risk between hazards categorised as high and intermediate risk at the Hazard Assessment Review Meetings support the qualitative categorisation of hazards into High, Intermediate and Low risk categories at these meetings.

86. The individual risk of premature fatality to a member of the public in the critical group from all of the assessed internal fault conditions is $2.4 \times 10^{-8}$ per year. The fault sequences which dominate are loss of primary filtration, dissolver scrubber failures, and cropping of short-cooled fuel.

87. The individual risk of premature fatality to a member of the public in the critical group from external events is $1.7 \times 10^{-9}$ per year. This risk is dominated by aircraft crash onto the Disassembly Cave with subsequent zinc bromide window failure, or on to the PuNit flask.

CONCLUSIONS

88. The record for the reprocessing plant has demonstrated a continued improvement in both operational and safety performance.

89. It has been shown that the plant and process is quite flexible in that modifications can be made to both in order to respond to circumstances.

90. It has been demonstrated that the plant and process has kept pace with PFRs fuel reprocessing requirements and is now operating routinely at close to ‘flowsheet’. This confirms the robustness of the original plant design.

91. The reprocessing of DFR and PFR irradiated fuel has been successfully subject to Safeguards for the last twenty-one years.

ACKNOWLEDGMENTS

92. The continued level of support of the UK Government is noted, as presently funded through the Department of Trade and Industry.

93. The positive and enthusiastic participation of all the operating, supervisory and management staff - together with the dedicated support of service team members - must be recognised. Without that it would not be possible to convey such an impressive record. The contribution to the project of members of the team who have left on retiral in recent years is not forgotten.
94. Finally, the encouragement of the Dounreay and UKAEA senior staff is acknowledged, particularly for those supporting this presentation.
BIBLIOGRAPHY


TABLE 1

PFR FUEL REPROCESSING

ANNUAL THROUGHPUT 1980 - 1994 (to date)

<table>
<thead>
<tr>
<th>Year - financial (nominal)</th>
<th>Reprocessing Plant Operating Days</th>
<th>Throughput Kg HM</th>
<th>Rate Kg/day</th>
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<tr>
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<tr>
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<td>109</td>
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<td>13.6</td>
</tr>
<tr>
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<td>44</td>
<td>538</td>
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<tr>
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<td>338</td>
<td>4788</td>
<td>14.2</td>
</tr>
<tr>
<td>Total/Average</td>
<td>1560</td>
<td>23143</td>
<td>14.9</td>
</tr>
</tbody>
</table>

HM ... Heavy Metal (U + Pu)
FAST REACTOR FUEL REPROCESSING BUILDING

1. IRRADIATED FUEL TRANSFER FLASK
2. FUEL ASSEMBLY BREAKDOWN CAVE
3. FUEL ASSEMBLY BREAKDOWN CAVE OPERATING FACE
4. FUEL DISSOLVER CELL
5. HIGH ACTIVE CELLS 1 & 2 SOLVENT EXTRACTION
6. MEDIUM ACTIVE CELL SOLVENT EXTRACTION
7. LOW ACTIVE CELL SOLVENT EXTRACTION
8. CONTROL ROOM
9. VENTILATION FILTER UNIT
10. WASTE REMOVAL FLASK
11. LASER ASSEMBLY
12. BREAKDOWN CAVE MAINTENANCE BOOTH
13. DECONTAMINATION BOXES
14. MANIPULATOR MAINTENANCE AREA
15. SAMPLE TANK ANNEXE
16. MIXER SETTLER PULSE UNITS
17. ANALYTICAL SAMPLE STATION
18. SOLVENT EXTRACTION CELLS ACCESS AREAS
19. VEHICLE ACCESS BAY
20. WASTE FLASK TRANSFER HATCH

Figure 2
DESIGN BASIS OF OFF-SITE EMERGENCY RESPONSE PLANS
FOR FUEL CYCLE INSTALLATIONS

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1. **Introduction**

In France, the term "off-site emergency response plan" (French abbreviation PPI) refers to all the arrangements which should be made by the government authorities to protect the population in the event of an accident affecting the installations of the site considered.

Management of the medium- and long-term consequences of the accident is covered in the post-accident action plan (French abbreviation PPA) which is intended to set out the provisions for a subsequent return to normal living conditions in the areas affected by the accident. It should also be borne in mind that management of the accident situation on site is covered in the on-site emergency response plan (French abbreviation PUI) implemented by the operator.

To determine the nature of the counter-measures which could be taken, it is necessary in the event of an accident to determine the upper bound case for releases of radioactive materials or chemicals. Several "typical accidents" have thus to be defined together with associated "source terms" representing releases which may arise from the different installations of the site.

The approach adopted in France consists in drawing up a list, as exhaustive as possible, of conceivable beyond-design basis accident situations which may affect the installations of the site considered.
These accident situations can be sought on the basis of reference accident situations dealt with in the safety reports of the installations involved. The most selected case consists in considering one or more aggravating factors with regard to these reference accidents (e.g. containment damage, filtration failure).

After identifying all the installations which could be at the origin of a significant release of radioactive or toxic products, the associated risks (fire, explosion, criticality etc.) are assessed as realistically as possible for each of them. This evaluation leads to the adoption of upper bound accident situations characterised in particular by a "release level" and kinetics derived from this assessment.

The "source terms" expressed in terms of quantities released with associated kinetics make it possible to calculate dose equivalents which could be received by members of the public or concentrations in toxic elements, according to the distance from the damaged installation.

The counter-measures to be provided in the off-site emergency response plan are based on the assessment on the one hand of the atmospheric dispersion of dangerous products corresponding to "source terms" and, on the other hand, of health effects which may arise, quantified on the basis of radiological and/or toxic data.

Based on these results, distances are suggested for the government authorities to take into account for the operational part of the off-site emergency response plan. The counter-measures put forward are essentially confinement and evacuation.

The distances relative to these two sorts of counter-measures are calculated in part on the basis of numerical values of intervention levels recommended by the International Commission on Radiological Protection (ICRP), in part on the basis of available toxicological data.

Two cases for the application of the methodology are put forward : one relates to front-end fuel cycle installations (Tricastin nuclear site), and the other to spent fuel reprocessing installations (COGEMA's La Hague plant).
2. Outline of the method

2.1. Definition of typical accidents

An identification of all the risks encountered in each installation leads to a consideration of the risks represented by the equipment and processes used, those represented by goods received, stored, used and produced, those provoked by human failure, and those provoked by events of an external origin.

In normal operation or during transients in the installations at the sites considered, the main probable accident situations are as follows:

- critical excursion,
- irradiation,
- dispersal of radioactive and/or toxic substances.

Possible sources of these risks include:

- process or equipment failure: fires, explosions, loss of radiation shielding, leakage from equipment, build-up of explosive gas or dust, dropped loads, modification to the sub-criticality parameters, pipe or valve rupture, corrosion, wear, flames or fire used during work;
- events of external origin: aircraft crashes, overpressure caused by explosion of an oil tanker, a barge, an explosive gas storage tank, a heating system or gas line. Any overpressure calculations are performed under the same atmospheric conditions as the estimate of the consequences of the resulting release;
- human failures.

It should be borne in mind that the approach adopted has to ensure a certain amount of realism, especially regarding the hypotheses to be taken into account when drawing up the accident scenarios; the nuclear operator's means of prevention, detection and action can be taken into account when estimating the duration of the release.
These accident scenarios can be drawn up on the basis of existing scenarios already dealt with in the safety reports and on-site emergency response plans (PUIs) for the installations in question. Indeed, the accidents set out in Level 3 of the on-site emergency response plans (accident situations which are radiological or toxicological in nature with off-site consequences) should be consistent with the typical accidents adopted in the technical bases for the off-site emergency response plans. Generally speaking, one or more aggravating factors are added to the design basis accidents set out in the safety reports, to wit damage, failure or additional loss of confinement.

Usually, an initial, mainly qualitative approach is used to dismiss, a priori, certain accident scenarios which do not appear to have significant off-site consequences or those with a very low probability. A more detailed analysis (quantitative) is then used to determine accidents with insignificant off-site consequences (accidents dismissed a posteriori) and to adopt upper bound accident situations with release levels and kinetics: the "source term".

2.2. Evaluation of the "source terms" associated with the accident situations considered

Mention is made of the hypotheses needed to assess the "source terms" (quantities involved, suspension coefficients in the installation, routing of substances through the installation, deposition coefficients, filtration coefficients).

- Quantities involved.

- Firstly, the quantities present in the unit in question are set in terms of an upper bound figure valid for any given moment (maximum authorised quantities, maximum vessel fill-up level).
- Secondly, the part (or number of units) affected by the accident is selected according to the type of confinement used to keep substances in the installation and the presence of high-risk equipment.
- Thirdly, the nature and physical and chemical properties of the substances involved are considered in such a manner as to estimate their leakage state (solid, gaseous or liquid) and any reactions they may undergo with the different environments they pass through (in particular UF6).
- Fourthly, the leak rates (single phase or two phase depending on the situation), the instantaneous vaporisation rate, pool evaporation rate and suspension factors are evaluated.
- In the meantime, the associated durations are estimated in the light of certain operator intervention equipment (bunds, remotely controlled valves, spreading of neutralising products, water curtains, emplacement of lids etc.).
- Height and release paths.

Radioactive and/or toxic material can migrate directly if their sole containment has been damaged (containers stored outside, for instance).

In all other cases, migration, filtration and deposition coefficients can be adopted to take account of the various containment barriers, passage through intermediate rooms (estimation of any involvement in a fire according to their heat load density), and filtration (estimate of the number of floors spared in the event of a fire).

Releases are characterised in terms of the height at which they are released into the open air. Natural leaks from the building (windows, doors) or releases at a height taking into account the height of stacks and any buildings in the vicinity are considered as ground-level releases.

Furthermore, a rise in the height of the plume resulting from the large amount of heat energy at the source during a fire can be taken into account when estimating the height of the release.

- Release kinetics.

An estimate of the kinetics (slow or fast) is essential when evaluating consequences and depends on the many hypotheses set out above.

The "source term" is therefore characterised by a quantity (activity or mass) of substances (physical and chemical nature, composition) released at a specific height above ground for a specific period.

2.3. Evaluation of health consequences

Public health consequences are expressed in terms of effective committed dose equivalents to the whole body and to the critical organ in the case of radiological consequences, and in terms of concentration couples of toxic elements/exposure periods (corresponding to the toxic load) in the case of toxicological consequences, which can be received by members of the public at different distances from the installation in question.

- Atmospheric dispersion.

The most critical contamination path to man in the short term is the atmospheric path. Atmospheric dispersion calculations are performed, taking into account adverse meteorological conditions to ensure that the counter-measures proposed for protecting the public cover the worst-case scenario.
The meteorological conditions chosen depend on the site in question so that any calculation of health consequences will remain realistic (dry weather and the Mistral wind in the Rhône valley for instance).

All calculations performed take account of Douyr atmospheric dissemination parameters using the codes developed at the Institute for Nuclear Safety and Protection (IPSN), adapting them to the different types of release (products, kinetic) and to the geometry of the source.

For releases lasting longer than thirty minutes, a corrective factor can be applied to the dispersion calculation to take account of the gusts of wind over time (variations in the direction of the wind for the duration of the release).

- The radiological viewpoint

In Publication 63, the International Commission on Radiological Protection (ICRP) recommended a certain degree of flexibility in ascribing numerical values to the intervention levels for applying measures to protect the public in the event of a nuclear accident.

In France, distances around the site in question where counter-measures must be put into effect as part of the off-site emergency response plan are evaluated for each of the "justified" intervention levels relating to evacuation and confinement counter-measures. The values of these levels, recommended by the ICRP, are:

<table>
<thead>
<tr>
<th></th>
<th>confinement</th>
<th>evacuation</th>
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</thead>
<tbody>
<tr>
<td>committed dose equivalent to</td>
<td>50 mSv</td>
<td>500 mSv</td>
</tr>
<tr>
<td>the whole body</td>
<td></td>
<td></td>
</tr>
<tr>
<td>committed dose equivalent to</td>
<td>500 mSv</td>
<td>5 Sv</td>
</tr>
<tr>
<td>the critical organ</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- Chemical toxicity

Two types of data are currently used for health effects following the dissemination of chemical products:

- toxicity graphs (of the DANDRES type for chlorine), the use of which is recommended by the Institute for Nuclear Safety and Protection. This is a compilation of charts showing the various levels at which toxic effects appear in humans (irritation, illness, danger, lethal) in terms of concentration in air and the length of exposure at this concentration. These curves were plotted
specifically for hydrofluoric acid, ammonia and chlorine. The concentration adopted for implementing counter-measures corresponds to the level at which the "illness" effect appears:

- IDLH. This is a bearable exposure level, for an exposure of thirty minutes, with no irreversible effect on the organism. These values were obtained from tests to qualify protective masks for workers. If the length of exposure is not equal to thirty minutes, the IDLH concentration is extrapolated back, taking account of the toxicity factor for the chemical in question when this is known.

- Uranium toxicity

By virtue of the chemical toxicity of uranium, its dissemination provokes certain toxicological consequences, in addition to the consequences arising out of its radioactive nature. The atmospheric concentration adopted when determining the distances at which counter-measures must be implemented to protect the population corresponds to the level at which the "renal lesion" effect arises.

In the case of accidents with both toxicological (chemical and/or due to uranium) and radiological consequences, the distances at which counter-measures must be taken to protect the population would be the largest distances calculated for one of the types of consequences.

The practical conditions and resources needed to implement counter-measures are to be laid down by the competent government authority. However, it may be recommended to take certain actions in this area, particularly in view of the short response times required for certain accident situations with fast kinetics: these are mainly precise local counter-measures, specific to the site, that the operator may have to implement on behalf of the government authorities before they actually intervene.
3. Application to installations from the front-end of the fuel cycle: the Tricastin site

The Tricastin site houses many installations from the front-end of the fuel cycle:

- the COMURHEX plant for converting uranium into uranium hexafluoride (UF₆),
- the EURODIF plant for the enrichment of uranium isotopes by gaseous diffusion,
- the FBFC plant for the fabrication of uranium-based fuel for pressurised water reactors,
- installations for uranium chemistry and for alternative uses of UF₆ run by the Compagnie Générale des Matières Nucléaires (COGEMA), including the W plant for the defluorination of uranium with a low concentration of the 235 isotope from the EURODIF plant and from the UF₆ transfer and sampling units,
- the SOCATRI plant for decontaminating equipment and for processing solutions containing uranium,
- installations run by the Commissariat à l'Énergie Atomique (CEA) carrying out research and development into enrichment processes.

One of the main features of all of these installations, from the point of view of risks, is the use of uranium with varying concentrations of the 235 isotope, especially in the form of UF₆, and various chemical reagents (especially hydrofluoric acid (both anhydrous and in solution, chlorine and ammonia).

As part of the approach outlined earlier, a tentative list of accident situations was drawn up on the basis of installation characteristics, especially from the point of view of confinement and the conditions in which the various products are used, whether chemical or radioactive, the quantities and physical and chemical forms of the products and any possible events which could arise out of the environment of the site.

The main accidents listed are as follows:

- criticality accident (generation of $5 \times 10^{18}$ fission reactions in 10 minutes),
- release of UF₆,
- explosion involving hydrogen arising in furnaces used for the production of products containing uranium,
- fire involving contaminated solvents or oils pending incineration,
- release of chlorine,
- release of chlorine trifluoride,
- release of ammonia,
- release of hydrofluoric acid.
As regards chemical reagents, account is taken of events which could affect storage installations, especially when filling storage tanks, either for feeding in reagents or for removing products after production, distribution installations and transport operations carried out on the site.

The main probable scenarios were drawn up for each of these accident situations and studied with a view to adopting typical accidents for the production of technical bases for performing the sizing calculations for the site off-site emergency response plan.

The following typical accidents were adopted for the purposes of the sizing calculations.

1. A leak of UF₆ from a cask holding natural UF₆ being cooled in an outdoor yard. This container contains the largest quantity of UF₆ in a form which can easily be spread outside a building (liquid form), the loss of the confinement function of such a container was assumed. To retain some degree of realism, it was decided to assume a leak from a broken valve.

2. An aircraft crashes onto a storage yard for casks containing UF₆ at ambient temperature. In view of the fuel fire likely to break out after the accident; a release of gaseous UF₆ into the atmosphere is adopted.

3. An aircraft crashes onto the EURODIF gaseous diffusion plant, followed by the fuel in the plant's fuel tanks catching fire; a release of gaseous UF₆ is assumed.

4. A leak of UF₆ from a cask, assumed to have been over-filled and to be undergoing heating in an oven. In view of the characteristics of the building housing the oven, this accident would result in different releases, depending on whether or not one accepts the hypothesis of redeposition of most of the uranium in the immediate vicinity (in or around the building), in the case of a leak of liquid UF₆.

5. A leak of anhydrous hydrofluoric acid from a distribution pipe from the cold store for this reagent or when filling a delivery tanker.

6. A leak affecting a store for a 70% solution of hydrofluoric acid attached to the W defluorination plant, following the loss of confinement of a vessel combined with the failure of devices for limiting existing consequences to act immediately.

7. A handling accident involving a container of one tonne of liquid chlorine resulting, following the rupture of a valve, resulting in a release of gaseous chlorine.
8. A leak at an ammonia tank following a pipe break, resulting in a gaseous release.

9. A leak when filling up an ammonia delivery tanker resulting in a gaseous release.

The environmental consequences of these scenarios were evaluated using the methodology set out above, taking into account the different adverse meteorological conditions at the various release heights adopted (at ground level for the majority of postulated accidents, except in the case of aircraft crashes where an increase in the height was assumed as a result of the heat given off by the fire).

It therefore became possible to determine the distances at which consequences are deemed intolerable in terms of the criteria adopted. These distances are in the region of:
- 4 km for releases of UF₆,
- 2 km for releases of other toxic substances.

The following can be concluded:
- the consequences of the above accident are chemical in nature, the radiological aspects are of only secondary importance in view of the physical-chemical form of the uranium and its enrichment,
- the accidents all display rapid kinetics. It is therefore the operator's responsibility to alert the population (by sirens); furthermore, the main probable counter-measure for protecting the population would be rapid confinement following the alert.
4. Application to installations from the back-end of the fuel cycle: the La Hague site

4.1. Typical accidents considered

4.1.1. Events occurring off-site

In the case of COGEMA's La Hague establishment, owing to the specific requirements of site isolation, it is not worthwhile adopting events occurring off-site for the technical bases which will be used to draw up the off-site emergency response plan.

4.1.2. Typical accidents associated with non-nuclear installations

The La Hague site houses the following installations:

- vessels for storing nitric acid and soda in solution (non-flammable toxic products),
- formaldehyde stores (toxic and flammable),
- reagent stores,
- fuel oil stores,
- propane stores,
- hydrogen store.

The consequences of accidents for each installation should not require the implementation of countermeasures to ensure the protection of nearby populations. The accidents associated with non-nuclear installations do not therefore have to be adopted when drawing up the technical bases for the off-site emergency response plan.

4.1.3. Critical excursion

The typical associated accident corresponds to $5 \times 10^{18}$ fission reactions in 10 minutes and would result in a "source-term" in the order of a few $10^{14}$ Bq, mainly consisting of rare gases and a little iodine. The radiological consequences of such a release off-site would be low, in the region of a few mSv at most, both in terms of external exposure to the whole body and in terms of equivalent dose committed to the thyroid by inhalation of iodine.
4.1.4 Typical accidents related to releases of fission products

a) The first postulated typical accident corresponds to the prolonged loss of cooling to storage vessels for concentrated solutions of fission products.

b) The second typical accident postulated consists in the failure of the open air scavenging system for a vessel for the storage of solutions with high concentrations of fission products which could, through the accumulation of hydrogen produced by radiolysis, result in an explosion whose consequences would include damage to the vessel itself and also destruction of the cooling system.

The consequences would not require the implementation of counter-measures to protect the population.

4.1.5 Typical accidents associated with releases of plutonium

Plutonium is present in the installations in various forms, particularly in the form of an aqueous or organic solution or as PuO₂ powder. In all cases, the risk to be borne in mind in the installations in question is the dispersion of plutonium following a fire.

Firstly, a solvent fire was adopted as the typical accident, because of the possibility of large quantities of plutonium being involved, also assuming the occurrence of certain containment failures which could lead to rapid releases off-site.

Secondly, it was considered that a fire could also be the root cause of the dispersion of PuO₂ powder.

4.2 Typical accidents adopted for drawing up the technical bases for the off-site emergency response plan for COGEMA's La Hague site

4.2.1 Solvent fire

In the most adverse meteorological conditions, dose equivalents committed to the whole body at the site perimeter, in the first homes (about 1000 m from the release point) and at various distances are such that:

- there is no need to evacuate the population for health reasons;
- populations should be confined to their homes for a distance of about 2600 meters around the accident-hit installation.
4.2.2. **Fire involving plutonium oxide powder**

Using considerations similar to those developed in the previous scenario, the corresponding radiological consequences are so insignificant that no counter-measures seem necessary to protect the health of the populations.

4.3. **Conclusion for the La Hague site**

The technical bases, updated after the UP3-A plant was commissioned, and supposed to serve as the basis for the off-site emergency response plan for the La Hague site, are taken from a study of possible accidents likely to affect site installations. The results of this study show that:

- there is absolutely no need to evacuate the population for health reasons,
- populations living within a radius of 2800 m of the site may have to be confined to their homes.

In view of the rapid kinetics of the release (one hour), the alert must be given immediately and is therefore the responsibility of the operator in accordance with conditions and resources to be decided on in collaboration with the government authorities.

Consequently, the 2 km and 5 km zones provided for in the current off-site emergency response plan for the evacuation and sheltering of populations, which were based on former accident studies carried out for the UP2-400 plant, are well in excess of the worst-case scenarios.
5. General conclusion

The process of drawing up off-site emergency response plans for nuclear sites is based on the identification of types of accidents which could arise in the installations and which could lead to releases of radioactive materials in such quantities that it could prove necessary to take measures to protect the population.

The safety provisions made are such that accidents of this sort are of course highly unlikely. The types of accidents considered when drawing up the off-site emergency response plans are therefore one-off situations which are only considered as an additional precaution for protecting the population.

Life-size emergency exercises with the functioning of emergency response centres and involving the authorities and staff are carried out regularly, based on a particular accident hypothesis.
IMPACT OF THE LA HAGUE REPROCESSING PLANTS ON THE SURROUNDING ENVIRONMENT

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COGEMA

1. INTRODUCTION

The La Hague UP2-800 and UP3 plants, designed to reprocess a total of 1600 metric tons per year of spent fuel, have been commissioned and are on the way to reach their nominal capacity.

One important condition for this operation is the radiological protection of the environment. Regulatory limits must be respected, and the impact on population must be as low as reasonably achievable, social and economic factors being taken into account.

2. REGULATORY REQUIREMENTS

Before start-up of a nuclear facility, the impacts of facility operations on the surrounding environment and, more specifically, on the public must be carefully assessed. This assessment addresses the health effects of plant operations under both normal and accidental conditions, and is the basis for official French authorities, which establish release limits to be respected at all times. The derived regulations concern liquid and gaseous releases; in the case of the La Hague reprocessing plant, they consist in four activity thresholds for specific elements, which the authority considered as being representative of the impact:

- liquid releases: tritium, beta emitters (excluding tritium), strontium 90 and cesium 137, and alpha emitters;
- gaseous releases: gases other than tritium, tritium, halogens, and aerosols.

<table>
<thead>
<tr>
<th>LIQUIDRELEASES</th>
<th>GASEOUS RELEASES</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tritium</td>
<td>37,000 TBq</td>
</tr>
<tr>
<td>Beta emitters (excluding tritium)</td>
<td>1,700 TBq</td>
</tr>
<tr>
<td>90 Sr and 137 Cs</td>
<td>220 TBq</td>
</tr>
<tr>
<td>Alpha emitters</td>
<td>1.7 TBq</td>
</tr>
<tr>
<td>Gases (other than tritium)</td>
<td>480,000 TBq</td>
</tr>
<tr>
<td>Tritium</td>
<td>2,200 TBq</td>
</tr>
<tr>
<td>Halogens</td>
<td>110 Gbq</td>
</tr>
<tr>
<td>Aerosols</td>
<td>74 Gbq</td>
</tr>
</tbody>
</table>
Compliance with the release limits is checked through radiological monitoring of the site and its environment. All results of radiological measurements which trace the evolution of the environment are sent to be checked to the Office de Protection contre les Rayonnements Ionisants (OPRI), or Radiation Protection Office, of the Ministry of Health. The OPRI cross-checks these results with the results of its own measurements.

3. THE MONITORING PROGRAM

The methods used for monitoring and surveillance may be divided into two different categories with respect to their implementation:

- continuous monitoring, which rapidly activates necessary corrective actions in the event of changes in a monitored parameter; due to the large number of parameters involved, monitoring is conducted from a central control room, the "PC environment" of La Hague.

- delayed monitoring, using environmental sampling and laboratory analysis, to add further data to that gathered by continuous monitoring and dosimetry.

Continuous Monitoring

Real-time monitoring is performed on gaseous releases, drainage systems and meteorological parameters.

Gaseous Releases

Gaseous releases are monitored continuously on three concentric circles starting with the plant. The inner circle centers on the main stacks of the UP2, UP2 800, and UP3 plants. Measurements are taken of alpha, beta activities and of gas flow-rates. The second circle is made up of eight monitoring stations on the perimeter of the plant site which help to determine if an off-normal event may have detrimental effects off site. Measurements are taken of the radioactivity of airborne elements and of ambient radiation.

The third circle is made up of five monitoring stations located in neighboring villages 1 to 5 kilometers from the site. Gas activity and radiation are measured in real time.

Drainage Systems

All water passing through the site is monitored before release to the surrounding environment by five beta monitors. The flow rate is continuously measured, and complementary chemical analysis of water are also performed.

Meteorological Parameters

A meteorological station is installed on site. It will help to determine transfers of radioactive gaseous effluent releases to the atmosphere during both normal and accidental operating conditions.
Central Control Room

Monitoring data are centralized to provide an immediate overall view of radiological conditions at the plant and in the environment. Each monitoring station can automatically develop and display the data locally, and send it to the central environmental control station. The central control station's multi-tasking computer receives data in real-time from the various monitoring stations, immediately displays them on color monitors, prints out data reports and archives data files.

Delayed Monitoring

Delayed monitoring involves the taking of representative samples from the environment following a regular and periodic program, and analyzing them in the Environmental Laboratory operated by the Radiation Protection Department. The environmental monitoring program enables detailed dose calculations to be established; it involves some 17,000 samples a year taken from the three pathways for radionuclide migration to the food or biological chain — atmospheric, hydrogeological, and marine — and around 50,000 analyses.

Atmospheric Pathways

Continuous real-time monitoring is supplemented by measurements taken on filters and activated-carbon traps at each plant outlet and at the monitoring stations on the site boundary and in the outlying areas. Monitoring includes potential fall-out in rainwater, vegetation, crops, milk and meats.

Hydrological Pathways

The 32 springs and streams originating near the plant are monitored and analyzed. Through a site network of 220 piezometers, the water table can be closely monitored. The district's drinking water is also regularly analyzed.

Marine Pathways

A 5 kilometer long submerged pipeline carries liquid releases from the plant out to the sea. The pipeline is regularly inspected. Two hundred kilometers of coastline, from Granville to Le Havre, are sampled, including water, sand, sediment, crustaceans, shellfish and plant-life which act as filters for radionuclides released into the sea. Deep sea sampling includes water, sand, sediments and fish. COGEMA asks the French Navy to perform the sampling, and participates to the studies on sea dispersion of radionuclides in the English Channel and the North Sea, which are undertaken by the Marine Radioecology Laboratory of the French Atomic Energy Commission (CEA).

The La Hague monitoring program, which is approved by the Ministry of Health, is a source of valuable data. Figure 1 shows environmental monitoring data for the last ten years, along with measurements taken in 1965 and 1966, the reference years for natural site conditions prior to the start-up of the plant.

The main results of monitoring (real-time and delayed) and the releases to atmosphere and sea are communicated to the public through interactive multimedia terminals, one situated in the Belvedere near the plant, and one in Cherbourg's Espace Communication, a communal place near Cherbourg's Town Hall. The data are updated everyday.
4. RESULTS OF MONITORING

**ASSESSMENT OF MEASUREMENTS IN THE ENVIRONMENT OF LA HAGUE**

NEGligible impact on the environment

![Graph showing artificial and natural activities](image)

**Figure 1**

The highest activity levels are also the oldest (at the back of the table), and relate to marine monitoring. Activity levels have decreased over the last ten years, while the quantities of reprocessed fuel continued to grow. This decline in activity reflects lower releases of activity to the sea (figure 2), resulting from higher separation performances of the process, improved management of liquid streams and a new chemical precipitation process on the final effluents.

Figure 1 also shows that, for the last few years, man-made radioactivity has remained at the same level as natural radioactivity (potassium 40 and beryllium 7) for limpets, and even less for other samples.
LIQUID RELEASE : BETA GAMMA

Figure 2
5. CONCLUSION

Safe management of spent fuel through reprocessing involves the application of a mature technology and professional resources in an integrated and regulated manner.

The objective is to control occupational and public exposure to ionizing radiation and to protect the environment in accordance with national regulations and international recommendations.

In furtherance of these objectives, it is clear that the operation of the La Hague reprocessing plants results in doses to the public which are much lower that unavoidable natural exposure variations. This decrease has been achieved even as the throughput of the plant has been multiplied by 4.

This is a consequence of a consistent program implemented by COGEMA, of reduction of releases within the full adherence to the ALARA principle, as part of the radiation protection program of COGEMA. These reductions in dose and release limits are a tribute to the effort undertaken by COGEMA over the last 15 years to safely manage the spent fuel and do more than comply with the legal requirements.
ACTIVITIES ON PUBLIC RELATIONS WITH RESPECT TO FUEL CYCLE SAFETY IN JAPAN

Kenkichi HIROSE
Director
Nuclear Materials Regulation Division
Nuclear Safety Bureau
Science and Technology Agency
ACTIVITIES ON PUBLIC RELATIONS WITH RESPECT TO FUEL CYCLE SAFETY IN JAPAN

Considering the need of disclosing nuclear safety information to the public, the regulatory agency plans to establish a regime in which information disclosure tasks can be smoothly and responsibly conducted. Information to be disclosed includes applications for construction approval, accident and trouble information, and transportation safety data.

(1) With increasing concern of the public over nuclear safety, it is important to make the political process of nuclear safety regulation clearer so that the public can gain a better understanding of nuclear safety.

(2) We are now preparing for the disclosure of the construction approval applications and the safety codes for facilities on which objections are being raised or suites are being filed; the nuclear fuel facility at Rokkasho-mura and the Monju reactor are great concerns of the public.

(3) At nuclear fuel facilities where no reporting duty is assigned by legislation, even small troubles are being reported to the regulatory agency and released to the press since April, 1994.
(4) A great amount of time and effort is required to promote disclosure of information. The required tasks include establishing criteria for information that cannot be disclosed in light of nuclear nonproliferation and safeguard of nuclear materials, investigating overseas approaches, studying past incidents, and negotiating with utilities.

(5) To promote public understanding, political decisions have to be made to conduct disclosure of information appropriately; expressing the opinions of the regulatory agency should be encouraged. Note that incomplete data provides adverse effect.

(6) A regime that manages information disclosure, of which political importance is increasing, should be established. In order to fully cope with new disclosure tasks, an organization that deals with the disclosure of nuclear safety information have to be set up within the regulatory agency.

(7) Guidelines for disclosure have to be drawn up quickly. We are now working on the disclosed versions of applications and the disclosure of safety regulation status.
Disclosure of regulatory information on nuclear fuel facilities

1. Basic policy on disclosure of information related to nuclear energy

The Atomic Energy Basic Law stipulates the three principles of independence, democracy and disclosure. Among them, the principle of disclosure reads "to disclose the fruits." This principle was initially set out at the time of legislation as a security clause for peaceful utilization to prevent any fruits of nuclear research from being kept as military secret and used as weapons. Subsequently, the clause has been applied so that such information related to safety that may draw high national interests is disclosed while a cautious attitude is taken when it comes to such information related to nuclear nonproliferation, protection of nuclear materials and protection of property.

2. Current status of disclosure of information related to nuclear energy

(1) The government established Public Archives in June 1973 in an effort to assist in diffusion and improved understanding of nuclear-related knowledge and aim at smooth promotion of the development and utilization of nuclear energy. At the same time, it has made available for review by the public the applications for reactor installation permit and the documents related to the safety of reprocessing facilities in the Public Archives and the National Diet Library.
The results of disclosure since the fiscal year 1993 are shown on the figure attached. For your information, the numbers of requests for disclosure in FY 1991 were nine for the Japan Atomic Energy Research Institute, three for the Power Reactor and Nuclear Fuel Development Corporation, six for Japan Nuclear Fuel Service Co., Ltd., one for Japan Nuclear Fuel Industries Co., Ltd. and seven for other processing manufacturers. In FY 1992, there were two applications for the Japan Atomic Energy Research Institute, five for the Power Reactor and Nuclear Fuel Development Corporation, four for Japan Nuclear Fuel Ltd. and two for other processing manufacturers.

(2) For those documents including the applications for permit to use nuclear fuel material that are deemed to be of low interests to the public, the submission has been decided according to the amount of information included related to commercial secrets, protection of nuclear materials and nuclear nonproliferation. Every time any specific request was made by a legislator for a document.

(3) In the meantime, as the administrative litigation on the nuclear fuel cycle facility at Rokkasho was underway, the plaintiffs and others also moved to object to and raise an administrative litigation on the licensing of the design and the construction method. In addition, when the approval of containers related to the transport of returned plutonium was questioned in December 1992, it was not only the protesters but the media who applied requests for disclosure of the container approval application.
(4) To cope with this move, it was committed to the protestor to disclose the first site construction applications for the waste management facility and the reprocessing facility of the Japan Nuclear Fuel Limited after deleting sensitive information, following the decision of a policy by the Nuclear Safety Bureau to adjust and sort such information related to the protection of nuclear materials, nonproliferation and commercial secrets and personal information when a specific request is made to disclose such documents as the application for reactor installation permit, the application for designated reprocessing operator and the application for reprocessing operation permit that had been disclosed in a proactive manner as well as the application for permit of use and the application for site construction approval.

Since March 4, 1994, following the completion of the preparation for the above-mentioned two documents, they have been available for review at the National Diet Library and the Public Archives of the Science and Technology Agency. The protestor was informed of the availability for lending.

3. Policy for future disclosure

To deepen the understanding and obtain the cooperation of the nation in the peaceful utilization of nuclear energy, it is assumed to prepare for smooth disclosure of the applications of site construction approval and the security provisions of such facilities that may attract high social interests. More specifically, work is expected to be done in cooperation with operators to remove the portions of the documents that may give rise to hindrance.
In the near-term, work is expected on the nuclear fuel cycle facility at Rokkasho and the prototype fast breeder reactor "Monju" that are keys to the country's nuclear fuel cycle program.
<table>
<thead>
<tr>
<th>Company/Mgt.</th>
<th>Title</th>
<th>Date</th>
</tr>
</thead>
<tbody>
<tr>
<td>Japan Nuclear Fuel Ltd.</td>
<td>Application for approval of alteration in the fabrication project of nuclear fuel material at Rokkasho uranium enrichment plant (partial revision)</td>
<td>June 1993</td>
</tr>
<tr>
<td>Power Reactor and Nuclear Fuel Development Corporation</td>
<td>Partial revision on the application for admission of altering the establishment of reprocessing facility</td>
<td>July 1993</td>
</tr>
<tr>
<td>Japan Atomic Energy Research Institute</td>
<td>Notice of alteration in the dismantlement notice of the nuclear reactor facility of nuclear ship of Japan Atomic Energy Research Institute</td>
<td>August 1993</td>
</tr>
<tr>
<td>Power Reactor and Nuclear Fuel Development Corporation</td>
<td>Application for approval of alteration in the fabrication project of nuclear fuel material for uranium enrichment prototype plant</td>
<td>November 1993</td>
</tr>
<tr>
<td>Japan Atomic Energy Research Institute</td>
<td>Application for approval of the underground disposal project of wastes by Japan Atomic Energy Research Institute Tokai branch</td>
<td>November 1993</td>
</tr>
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<td>Japan Atomic Energy Research Institute</td>
<td>Notice of alteration in the dismantlement notice of the nuclear reactor facility of nuclear ship of Japan Atomic Energy Research Institute</td>
<td>November 1993</td>
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<tr>
<td>Power Research and Nuclear Fuel Development Corporation</td>
<td>Application for approval of alteration in the fabrication project of nuclear fuel material for uranium enrichment prototype plant</td>
<td>November 1993</td>
</tr>
<tr>
<td>Musashi Institute of Technology</td>
<td>Application for approval of alteration in the nuclear reactor establishment (alteration in the storage facility for nuclear fuel material)</td>
<td>December 1993</td>
</tr>
<tr>
<td>Japan Atomic Energy Research Institute</td>
<td>Application for approval of alteration in the nuclear reactor establishment by Japan Atomic Energy Research Institute (alteration in HSRR nuclear reactor facility)</td>
<td>December 1993</td>
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<td>Organization</td>
<td>Description</td>
<td>Date</td>
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<td>Japan Nuclear Fuel Ltd.</td>
<td>Documents attached to the application for approval of the design and</td>
<td>March 1994</td>
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<tr>
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<td>construction methods for the reprocessing plant at Rokkasho reprocessing</td>
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<td>First application for approval of the design and construction methods</td>
<td>March 1994</td>
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<td>for the reprocessing plant at Rokkasho reprocessing and waste disposal</td>
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<td>Japan Nuclear Fuel Ltd.</td>
<td>Partial revision on the text and attached documents of the application for</td>
<td>March 1994</td>
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<td>approval of the design and construction methods for the reprocessing plant</td>
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<td>at Rokkasho reprocessing and waste disposal facilities</td>
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<td>Japan Atomic Energy Research</td>
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<td>March 1994</td>
</tr>
<tr>
<td>Institute</td>
<td>facility at Japan Atomic Energy Research Institute Tokai branch</td>
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<td>Partial revision on the text and attached documents of the application for</td>
<td>March 1994</td>
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<td>March 1994</td>
</tr>
<tr>
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<td>Power Research and Nuclear</td>
<td>Application for approval of alteration in the nuclear reactor establishment</td>
<td>March 1994</td>
</tr>
<tr>
<td>Fuel Development Corporation</td>
<td>at Oarai Engineering Center</td>
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</tr>
<tr>
<td>Japan Nuclear Fuel Co.</td>
<td>Application for approval of alteration in the fabrication project of nuclear</td>
<td>April 1994</td>
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<td>April 1994</td>
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<tr>
<td>Nuclear Fuel Industries, Ltd.</td>
<td>Application for approval of alteration in the fabrication project of nuclear fuel material (Tokai branch)</td>
<td>April 1994</td>
</tr>
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<td>Partial revision on the application for approval of alteration in the fabrication project of nuclear fuel material (Tokai branch)</td>
<td>April 1994</td>
</tr>
<tr>
<td>Nuclear Fuel Industries, Ltd.</td>
<td>Application for approval of alteration in the fabrication project of nuclear fuel material (Kumatori branch)</td>
<td>April 1994</td>
</tr>
<tr>
<td>Nuclear Fuel Industries, Ltd.</td>
<td>Partial revision on the application for approval of alteration in the fabrication project of nuclear fuel material (Kumatori branch)</td>
<td>April 1994</td>
</tr>
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<td>Mitsubishi Nuclear Fuel Co., Ltd.</td>
<td>Application for approval of alteration in the fabrication project of nuclear fuel material</td>
<td>April 1994</td>
</tr>
<tr>
<td>Mitsubishi Nuclear Fuel Co., Ltd.</td>
<td>Partial revision on the application for approval of alteration in the fabrication project of nuclear fuel material</td>
<td>April 1994</td>
</tr>
<tr>
<td>Japan Nuclear Fuel Conversion Co., Ltd.</td>
<td>Application for approval of alteration in the fabrication project of nuclear fuel material</td>
<td>April 1994</td>
</tr>
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<td>Partial revision on the application for approval of alteration in the fabrication project of nuclear fuel material</td>
<td>April 1994</td>
</tr>
<tr>
<td>Japan Atomic Energy Research Institute</td>
<td>Partial revision on the text and attached documents of the application for approval of the underground disposal project of wastes by Japan Atomic Energy Research Institute Tokai branch</td>
<td>May 1994</td>
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The status of the suit filed against the Rokkasho-mura nuclear fuel cycle facility

Suit has been filed against administrative permission and designation regarding the Rokkasho-mura nuclear fuel cycle facility. Local people near the facility have filed the suit, demanding the cancellation of the administrative disposition. The case is now under trial in Aomori District Court, whose jurisdiction includes the facility site.

The administrative permission and designation of Rokkasho-mura nuclear fuel cycle facility were provided for the applicant by Prime Minister based on the provisions in "The law for the regulation of nuclear source materials, nuclear fuel materials and reactors." Two months after the disposition, an objection was raised based on "The Complaints Against Administrative Acts Inquires Act ", claiming that the administrative disposition was illegal both in procedure and content. Three months later, an action for cancellation or action for nullity or both were brought based on "The Administrative Litigation Act."

Four facilities in the Rokkasho-mura nuclear fuel cycle facility are now on trial. The first suit was filed in 1989 against the uranium enrichment plant, which was licenced in 1988. Next suit was filed in 1991 against the low-level radioactive waste storing center, followed by the waste management facility and the reprocessing plant in 1993.

The main trial steps of the Rokkasho-mura uranium enrichment plant, whose trial is most proceeding, are listed below.
August, 1988 : Licence was issued for processing business.

October, 1988 : Objection was raised.

July, 1989 : Suit was filed.

September, 1989 : Start of oral pleadings.

October, 1993 : Field verification was conducted.

Further, witness questioning will be held.

As for the Rokkasho-mura reprocessing plant, a few oral pleading sessions up to now have been held. A number of oral pleading sessions are expected in future because the claims of the plaintiff cover various aspects including the illegality of procedure and content of administrative disposition.

The defendant is determined to win the suit by asserting its appropriateness and rightness at every opportunity.
Administrative Coordinator of Liaison and Publicity on Nuclear Energy

1. Background of Establishing the Post of Administrative Coordinator of Liaison and Publicity on Nuclear Energy

In 1970 and 1971, some nuclear reactors started operation: Kansai Electric Power Co.'s Mihama-1, Japan Atomic Power Co. (Japco)'s Tsuruga-1 and Tokyo Electric Power Co.'s Fukushima 1-1. With the successive commission of new reactors, communication between the local communities hosting the plants and the Government has increased rapidly: that is, host localities wanted to submit more demands to the Government, while the Government needed to offer more information to them. Therefore, loud cries have arisen from the communities for opening a branch office of the central government in a host region, which led to the establishment of the office of administrative coordinator of liaison and publicity on nuclear energy in 1972. The offices are now 13 in number.

2. Activities of Administrative Coordinator of Liaison and Publicity on Nuclear Energy

Administrative coordinator of liaison and publicity on nuclear energy take charge of the following activities to enhance the understanding and to gain the confidence of the hosting communities of nuclear plants.
(1) Service as Channels of Communication between Hosting Local Communities and the Central Government

--Offering information and explanation to host local communities about the policies and the measures of the nation on nuclear energy
--Conveying the requests and the demands of localities to the central government

(2) Cooperation with local communities in various activities

--Planning and management of lectures on administrative policies on nuclear energy
--Cooperation with the local governments in developing publicity on nuclear energy
--Cooperation with the local governments in keeping surveillance on environmental radioactivity
--Cooperation in practicing a training to prevent disasters in the accidents caused by nuclear energy

(3) Releasing Information about Accidents and Troubles

About the accidents and the troubles in other regions, necessary information are offered to relevant agencies as the need arises.

For the above-mentioned activities, the Aomori Office of Atomic Energy Policy and Coordination covers JNFL's nuclear fuel cycle facilities in Rokkasho-mura, and the Mito Atomic Energy Office covers the nuclear fuel facilities in Tokai and Oarai areas.
List of Local Offices of Atomic Energy Policy and Coordination

1. Hokkaido Office of Atomic Energy Policy and Coordination
   191-7, Usubetsu, Kayanumamura, Tomari-mura, Furu-ku, Hokkaido
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   TEL 0135-75-3547
   FAX 0135-75-2760

2. Aomori Office of Atomic Energy Policy and Coordination
   2-4-1, Shin-machi, Aomori-shi, Aomori
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   TEL 0177-75-1910
   FAX 0177-75-4552

3. Miyagi Office of Atomic Energy Policy and Coordination
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   135, Onagawa, Onagawa-ku, Onagawa-cho, Oshika-gun, Miyagi
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   TEL 0225-54-8608
   FAX 0225-53-2184

   189, Ohno, Shimono-go, Ohkuma-machi, Futaba-gun, Fukushima
   〒979-13
   TEL 0240-32-2680
   FAX 0240-32-3388

5. Mito Atomic Energy Office
   4-1, Atago-machi, Mito-shi, Ibaragi
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   FAX 0292-31-3789

   28-23, Tanaka, Kashivazaki-shi, Niigata
   〒945
   TEL 0257-24-4041
   FAX 0257-21-3376

7. Ishikawa Office of Atomic Energy Policy and Coordination
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   FAX 0767-32-4710
8. Fukui Office of Atomic Energy Policy and Coordination
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   5814-19, Ikeshinden, Hamakita-cho, Ogasa-gun, Shizuoka
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   FAX 0537-85-2877

10. Shimane Office of Atomic Energy Policy and Coordination
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11. Ehime Office of Atomic Energy Policy and Coordination
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    FAX 0894-38-2208

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    FAX 0955-74-3492

    4-24, Nakaba-cho, Sendai-shi, Kagoshima
    ☎ 895
    TEL 0996-23-1982
    FAX 0996-25-4308

Science and Technology Agency
Office of Planning of Nuclear Fuel Cycle Facility Siting
2-2-1, Kasumigaseki, Chiyoda-ku, Tokyo
☎ 100
TEL 03-3581-5716
FAX 03-3581-0774
Local Offices and Nuclear Facilities

1. Hokkaido Office:
   Hokkaido Electric Power Co. -- Tomari Power Station
2. Aomori Office:
   Japan Nuclear Fuel Limited -- Nuclear Fuel Cycle Facilities
3. Miyagi Office:
   Tohoku Electric Power Co. -- Onagawa Nuclear Power Station
4. Fukushima Office:
   Tokyo Electric Power Co. -- Fukushima-I Nuclear Power Station
   Tokyo Electric Power Co. -- Fukushima-II Nuclear Power Station
5. Mito Office:
   Japan Atomic Energy Research Institute and Power Reactor and Nuclear
   Fuel Development Corporation -- Research and Development Facilities
   Japan Atomic Power Co. -- Tokai Nuclear Power Station
   -- Tokai-II Nuclear Power Station
6. Niigata Office:
   Tokyo Electric Power Co. -- Kashivazaki-Kariwa Nuclear Power Station
   Tohoku Electric Power Co. -- Maki Nuclear Power Station
7. Ishikawa Office:
   Hokuriku Electric Power Co. -- Shika Nuclear Power Station
8. Fukui Office:
   Japan Atomic Power Co. -- Tsuruga Nuclear Power Station
   Power Reactor and Nuclear Fuel Development Corporation --Fugen
   Power Reactor and Nuclear Fuel Development Corporation --Monju
   Kansai Electric Power Co. -- Mihama Power Station
   Kansai Electric Power Co. -- Ohi Power Station
   Kansai Electric Power Co. -- Takahama Power Station
9. Shizuoka Office:
   Chubu Electric Power Co. -- Hamaoka Nuclear Power Station
10. Shimane Office:
    Chugoku Electric Power Co. -- Shimane Nuclear Power Station
11. Ehime Office:
    Shikoku Electric Power Co. -- Ikata Power Station
12. Saga Office:
    Kyushu Electric Power Co. -- Genkai Nuclear Power Station
13. Kagoshima Office:
    Kyushu Electric Power Co. -- Sendai Nuclear Power Station