IAEA-TECDOC-419

SPENT FUEL MANAGEMENT: CURRENT STATUS AND PROSPECTS OF THE IAEA PROGRAMME

PROCEEDINGS OF AN ADVISORY GROUP MEETING
ON SPENT FUEL MANAGEMENT
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN VIENNA, 11–13 MARCH 1986

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FOREWORD

In order to provide successful and harmonious development of nuclear power, all links of the fuel cycle - obtaining, using, storing, reprocessing and disposing of nuclear material used in the operation of nuclear reactors - should be taken into consideration. However, emphasis can be shifted from one problem to another, depending on a number of subjective and objective reasons. The subjective reasons are dictated, as a rule, by the national situtation - limited natural resources, level of industrial development, socio-economical and political conditions, commercial interest, etc. The objective reasons reflect the status of the world's nuclear fuel cycle and the modern trends in nuclear power development.

Spent Fuel Management always was, maybe, not the most acute problem (because there was sufficient time reserve for its solution), but undoubtedly - one of the most complicated and most sensitive in series of technical, political and public problems of world Nuclear Fuel Cycle. It is also worth mentioning that because of the inability to solve this problem the Nuclear Power Programme in Austria was cancelled.

It would not be an exageration to say that today the accents in Nuclear Fuel Cycle are shifted appreciably to Spent Fuel Management, and this is the case when objective and subjective reasons are in good agreement in the majority of the countries.

The main peculiarities of the present situation which force to be active and consistent in the solution of important technical, economical and political issues dealing with Spent Fuel Management tactics and strategy, are briefly described as follows:

(i) at-reactor pools constructed for the short-term storage of water-cooled reactor spent fuel are overfilled with irradiated assemblies in overwhelming majority of countries having developed nuclear power;

(ii) in spite of the optimistic prognoses in the 60s and early 70s, reprocessing does not play an essential role in the world's spent fuel management policy, and it appears as if it will not seriously influence the amount of stored fuel in the next 10 - 15 years;

(iii) in all the world, there is not one spent fuel repository operating and as a matter of fact, the door is open for alternatives to existing spent fuel disposal approaches (e.g., Swedish, Swiss concept, etc.);

(iv) although the situation with uranium now gives no reason for trouble, more and more countries are looking for means of recycling uranium and plutonium;

(v) the alarming condition after the last accident on a nuclear power plant, undoubtedly, will create additional complications in the back-end of the nuclear fuel cycle. At least, it can lead to the stengthening of safety requirements and, as a consequence - to the partial revision of existing technical concepts of spent fuel storage, transportation, treatment and disposal.

Even a simple analysis of the above formulated theses, convincingly shows that there is a great deal of actual subjects for close international discussions, and in some cases, only cooperative efforts could assist in the optimal solution of vital spent fuel management issues, and allow the prevention of former mistakes in the future.

This report, which is a result of the second IAEA Advisory Group Meeting (the first was held in 1984), is intended to provide the reader with an overview of the status of Spent Fuel Management Programmes in a number of leading countries, with a description of the past IAEA activities in this field of Nuclear Fuel Cycle and with the Agency's plans for the next 2-3 years, based on the proposals of Member States.

The Agency wishes to thank all participants of the Advisory Group Meeting on Spent Fuel Management (11-13 March 1986, Vienna) for their fruitful contributions and, especially, the Chairman, Mr. J.P. Colton. The officer of the IAEA, responsible for the organization of the meeting and for editing the document was Mr. F. Sokolov of the Nuclear Materials and Fuel Cycle Technology Section.

CONTENTS

PART I — PRESENTATIONS OF THE IAEA SECRETARIAT

PART II — COUNTRY STATUS REPORTS

PART III — MAIN RESULTS OF THE ADVISORY GROUP MEETING

PART I
PRESENTATIONS OF THE IAEA SECRETARIAT

INTRODUCTION

At the end of 1985, there were 374 power reactors connected to electricity supply networks in 26 countries, producing 248, 577 megawatts of electrical power. During the year, 31 reactors accounting for a total of 29,152 megawatts of electrical generating capacity were newly connected to the grid. Construction was started on six more. At the end of 1985 more than 30,000 metric tonnes of spent water reactor fuel had been discharged worldwide from nuclear reactors. Only a small fraction of this fuel has been reprocessed. Much effort has been devoted to the development of technology of spent fuel storage, both wet and dry, and of spent fuel direct disposal.

The thirty thousand tonnes of discharged fuel is an increase of fifty percent over that reported in 1983 (20,000 tonnes). Projections at the present time indicate that as much as 200,000 tonnes of spent fuel will be generated by the year 2000. Interim storage of spent fuel is expected to dominate the back-end of the fuel cycle until early next century. Most countries using nuclear energy to produce electrical power in Europe and Japan have expressed their intention to reprocess their spent fuel and use the extracted fissile material in FBR programs. Excess plutonium is being used in some countries for thermal reactor recycle. There are other countries which have selected the once-through option for an indefinite period of time.

The final resolution in the long term is not yet clear and therefore the uncertainties in the back-end of the fuel cycle remain of crucial importance. The range of possible spent fuel management strategies - to reprocess, to defer reprocessing, or to use a once-through option- makes it difficult to project requirements for storage capacity and reprocessing needs. With the lack of precise directions, many countries continue to explore various options for handling the spent fuel (wet-dry, short-long term storage, consolidation, reprocessing, etc.).

Whichever option chosen, there appears to be no technical problem in designing, constructing, and operating the technology and facilities. The progress in the development of the various technologies indicates that many choices are available. Due to the significant costs related to research and development it is necessary that international cooperation continues to assist countries with small nuclear programs.

Both storage and reprocessing of the spent fuel have been demonstrated in some countries. Many studies of the technical feasibility of the permanent disposal of spent fuel as a waste have been made and while no demonstration has yet been made it appears that this technology also exists.

Spent fuel management is closely connected with other important programs that are being undertaken, both nationally and internationally. In particular, nuclear power generation, nuclear waste management, transport, and safety of nuclear fuel cycle facilities.

OBJECTIVE OF ADVISORY GROUP

The objectives of the Advisory Group include the following:

- Provide technical advise to the Secretariat regarding the Agency's program in the back-end of the Nuclear Fuel Cycle.
- Serve as a means of exchanging information on the current situation and progress of national programs.
- Discuss the Agency's publications in this field and to advise the Agency on proposals for reviews, publications, guidelines and studies.
- Assist in the coordination of international activity in the field of the back-end of the fuel cycle.

The program topics discussed during the Advisory Group Meeting included:

- A review of the present activities of the Agency in the area of the back-end of fuel cycle.
- Discussions of the participants' presentations on national current situation and future plans.
- A review of the IAEA's activities concluded since the last Advisory Group Meeting.
- Recommendations for future IAEA meetings and other activities in the area of the back-end of the fuel cycle.

IAEA SPENT FUEL MANAGEMENT PROGRAMME — PAST AND PRESENT

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The main objectives and strategies of the Agency activities in the area of Nuclear Fuel Cycle are to promote the exchange of information between Member States on technical, environmental and economic aspects of nuclear fuel cycle technology, to provide assistance to Member States in the planning, implementation and operation of nuclear fuel cycle facilities and to assist in the development of nuclear fuel cycle technology.

The aim of the IAEA spent fuel management sub-programme is to provide interested Member States with a forum for exchanging information on technical and economic aspects of spent fuel management, including spent fuel storage, transportation, reprocessing and recycling of fissile materials. In 1979 the IAEA identified interim storage of spent fuel as an important independent step in the nucler fuel cycle. One of the first studies of the Agency that pointed out the importance of spent fuel storage was a Working Group Review called the Regional Fuel Cycle Centre Study (RFCC) in 1977. In 1978 a special consultants group was convened by the Agency to identify in what the Agency should be involved and what was the status of spent fuel storage throughout the world. In the same year a joint IAEA/NEA program on the behaviour of spent fuel assemblies during extended storage (BEFAST) was initiated to study the behaviour of spent fuel and pool components during extended storage; a coordinated research programme was established by the Agency in 1981. In 1979, hosting the INFCE Study, the Agency supplied secretariat services to Working Group 6 with Spain and Argentina providing the co-chairmen. Also in 1979 the Agecy initiated a study on international spent fuel management. Experts from 24 countries and 3 international organizations took part in the study, which was completed in 1982. In November 1980 the IAEA organized, in cooperation with USDOE, a Specialists' Meeting on Spent Fuel Storage Alternatives. In 1983 the Agency and the Junta de Energia Nuclear of Spain organized a Seminar in Madrid on the technical and environmental aspects of spent fuel management. A Technical Committee Meeting on

Methods Used in the Design of Dry and Wet SFS Facilities was held in 1985 in Espoo, Finland.

As a result of the large number of meetings held during this time, a number of important documents was published. These included: 1) The Survey of World Experience in Storage of Water Reactor Spent Fuel in Water Pools (1982); 2) The Guidebook on Spent Fuel Storage (1984); 3) The Glossary of Terms related to SFS (1985); Three TECDOCs were released, namely on "Status of Spent Fuel Dry Storage Concepts" and on Status of LWR Rod Consolidation for Storage Purposes" also the Glossary (English version). The Russian version of the Glossary was discussed and reported at the Consultants' Meeting held in Vienna and will be released in 1986. A questionnaire on Dry and Updated Wet Storage Experience was distributed among Member States in 1985 and the first Consultants' meeting on preparation of this Survey Report was held in the UK in February 1986. The Survey is planned for publication as a IAEA Technical Report Series document in 1987. The BEFAST (Behaviour of Spent Fuel Assemblies during Extended Storage) Co-ordinated Research Programme was continued and important information concerning the integrity of fuel cladding and operational reliability of storage facility was obtained and analysed. The final draft was prepared, discussed during the Leningrad meeting (May 1986) and released as IAEA TECDOC-418 early 1987. In answer to the recommendations of program participants, the co-ordinated program will be continued in 1986-91.

A TECDOC was produced as a recommendation of the consultant group meeting on "Status of Treatment of LWR Fuel". A consultants' meeting was convened in June 1985 to survey the development status of construction materials for equipment. It was ageed that this subject is of great interest to Member States having nuclear fuel cycle facilities and it was recommended to hold a Technical Committee Meeting on "Materials Experience in the Back-End of the Nuclear Fuel Cycle" in September 1986.

Another part of SFM is the treatment of spent fuel. The Agency has limited its activities in reprocessing particularly due to the sensitivity of parts of the technology as it relates to non-proliferation issues. Due to the interest of some Agency members in general aspects of spent fuel treatment, the Secretariat monitors experience in the area and to a certain extent other related activities.

The Agency is currently looking at the potential use of noble metals (Tc, Cs and Sr) in the twenty-first century.

The Agency's program for 1987-88 will focus on three different areas:
1) Evaluation of information on spent fuel arisings and capacity requirements;
2) The enhancement of international co-operation in the selections of spent fuel storage options and practices;
3) The exchange of information on spent fuel treatment, the recycling of fissile materials and the recovery and utilization of other valuable elements.

IAEA SPENT FUEL MANAGEMENT PROGRAMME

SUMMARY OF ACTIVITIES
1976 - 1985

1976-78	Regional Fuel Cycle Center Study (RFCC) Subsections: Spent fuel storage, transportation, reprocessing
1977	IAEA/USNRC Workshop on Safety of Spent Fuel Storage, Washington, D.C.
1978	Special consultants' group: recommendations that IAEA should be involved in monitoring the spent fuel status throughout the world.
1978	Joint IAEA, OECD(NEA), Spain sponsored International Conference on Spent Fuel Storage, Madrid
1979-81	International Spent Fuel Management Working Group Meetings
1979-81	International Plutonium Storage Working Group Meetings
1978-80	Joint IAEA/NEA programme - BEFAST started: The study of the behaviour of spent fuel and pool components during extended storage in water-filled pools. Two phases: 1. world survey and 2. continuing surveillance programme.
1979-80	International Nuclear Fuel Cycle Coordination Study (INFCE): Working Group 6: Spent Fuel Storage; Working Group 4: Spent Fuel Reprocessing.
1981	Coordinated Research Programme on BEFAST initiated. IAEA has signed coordinated research agreements with 11 institutes.

1981	Spent Fuel Storage alternatives meeting IAEA/USDOE, Las Vegas
1982	Storage of Water Reactor Spent Fuel in Water Pools - Survey of World Experience
1982	Consultants' group on Spent Fuel Reprocessing, Vienna
1983	Seminar on Technical and Environmental Aspects of Spent Fuel Management, IAEA/Spain, Madrid
1983	Nuclear Fuel Cycle Information System (NFCIS) initiated filing system
1984	OECD/NEA-IAEA Seminar on Remote Handling Equipment for Nuclear Fuel Cycle Facilities, Harwell, UK
1984	Advisory Group Meeting on Spent Fuel Management, Vienna
1984	Guidebook on Spent Fuel Storage - Summary of IAEA activities in Spent Fuel Storage (1976-1983)
1985	Technical Document "Status of the Treatment of Irradiated LWR Fuel (reprocessing status report)
1985	Glossary of Terms for Wet and Dry Storage of Spent Fuel
1985	Review of meetings and bibliography on topics related to Spent Fuel Management
1985	Technical Document "Status of Spent Fuel dry storage concepts: Concerns, issues and developments"
1985	Technical Document "Status of LWR rod consolidation for storage purposes: Concerns, issues and trends"
1985	Technical Document "Methods used in design of spent fuel facilities", Espoo, Finland (proceedings published)

Meetings on topics

on SPENT FUEL MANAGEMENT

1980 - 1986

/ not including meetings with emphasis on waste management only/

No.	Title	Place	Date
1	Annual meeting on nuclear technology (Reactor Conference '80)	Berlin	25-27 May 1980
2	A N S Annual meeting	Las Vegas, USA	8-13 June 1980
3	89 Annual meeting of the American Institute of Chemical Engineers	Portland, USA	17-20 Aug.1980
4	Directions in nuclear engineering research conference	Cambridge, UK	19 Sept.1980
5	National topical meeting of fuel cycles for the 80's	Gatlinburg, USA	29 Sept.-2 Oct. 1980
6	Annual meeting of Deutscher Kalte-und Klimatechnischer Verein	Berlin	15-17 Oct.1980
7	IAEA Advisory Group/Specialists meeting on spent fuel storage alternatives	Las Vegas, USA	17-21 Nov. 1980
8	3rd Annual meeting of the Materials Research Society	Boston, USA	17-20 Nov. 1980
9	2nd Annual national waste terminal storage (NWTS) information meeting	Columbus, USA	9-11 Dec. 1980
10	Meeting on the nuclear fuel cycle	Saclay, France	5-13 May 1981
11	11.meeting - how to achieve technical reliability under conditions relating to the future	Nuremberg, Germany,FR	13-15 May 1981

No.	Title	Place	Date
12	Fuel cycle conference	Los Angeles, USA	15-18 May 1981
13	Annual meeting on nuclear technology '81	Dusseldorf, Germany,FR	24-26 May 1981
14	A N S annual meeting	Miami Beach, USA	7-12 June 1981
15	A N S technical base for nuclear fuel cycle policy	Newport, USA	20-23 Sept.1981
16	Autumn meeting on metallurgy	Paris, France	20-22 Oct. 1981
17	International study meeting on modern power stations	Liège, Belgium	26-30 Oct.1981
18	Annual meeting of the Institute of Chemical Engieneers	New Orleans, USA	8-12 Nov.1981
19	International conference of fast reactor fuel cycles	London, UK	9-12 Nov.1981
20	N W T S program information meeting	Columbus, USA	17-19 Nov.1981
21	A N S winter meeting	San Francisco, USA	29 Nov.-4 Dec. 1981
22	Meeting on LWR extended burnup fuel performance and utilization	Williamsburg, USA	4-8 April 1982
23	A N S topical meeting on treatment and handling of radio-active wastes	Richland, USA	19-22 April 1982
24	International ENS/ANS conference on nuclear energy with emphasis of fuel cycles	Brussels, Belgium	26-30 Apr. 1982

No.	Title	Place	Date
25	OECD specialist workshop on techniques for the dry storage of spent fuel elements	Madrid, Spain	11-13 May 1982
26	American Nuclear Society - annual meeting	Los Angeles, USA	6-11 June 1982
27	26 American Association of Cost Engineers- Annual meeting	Houston, USA	27-30 June 1982
28	International topical meeting on liquid metal fast breeder reactor safety and related design and operational aspects	Lyon, France	19-23 July 1982
29	Institute of Nuclear Materials meeting	Washington, USA	19-21 July 1982
30	International conference on nuclear power experience	Vienna, Austria	13-17 Sept.1982
31	Symposium on gas-cooled reactors today	Bristol, UK	20-24 Sept.1982
32	NRC meeting on advances in reactor physics and core thermal hydraulics	Kiamesha Lake, USA	22-24 Sept.1982
33	ANS topical meeting - spent fuel storage	Savannah, USA	26-29 Sept.1982
34	10. Water reactor safety research information conference	Gaithersburg, USA	12-15 Oct.1982
35	A N S winter meeting	Washington, USA	14-19 Nov.1982
36	Symposium on remote sensing technolgoy	Las Vegas, USA	23-25 Febr.1983
37	C S N I specialists meeting on interaction of fire and explosion with ventilation systems in nuclear facilities	Los Alamos, USA	25-28 April 1983

No.	Title	Place	Date
38	American Ceramic Society - annual meeting	Chicago, USA	25-28 April 1983
39	7th International symposium on packaging and transportation of radioactive materials	New Orleans, USA	15-20 May 1983
40	General meeting 83 of Deutsche Gesellschaft fur Metallkunde e.V.	Erlangen, Germany,FR	24-27 May 1983
41	A N S Annual meeting	Detroit, USA	12-17 June 1983
42	24 Annual meeting of the Institute of Nuclear Materials Management	Vail, USA	10-13 July 1983
43	N R C workshop on spent fuel/cladding reactions during dry storage	Gaithersburg, USA	17-18 Aug.1983
44	Joint power generation conference	Indianapolis, USA	25-29 Sept.1983
45	IAEA Seminar on technical and environmental aspects of spent fuel management	Madrid, Spain	27-30 Sept.1983
46	11. NRC water reactor safety research information meeting	Gaithersburg, USA	14-24 Oct.1983
47	A N S winter meeting	San Francisco, USA	30 Oct.-4 Nov.83
48	Meeting of fuel reprocessing	Paris, France	8 Nov. 1983
49	Topical meeting on safeguards technology - the process safeguards interface	Hilton, Head Island, USA	28 Nov.-1 Dec.83
50	Meeting on protection, handling, detection and safety (PMDS 83)	Saclay, France	6-7 Dec.1983
51	INMM Seminar on spent fuel storage	Washington, USA	10-13Jan.1984

No.	Title	Place	Date
52	IAEA expert meeting on spent fuel management	Vienna, Austria	27-30 May 1984
53	INMM packaging and transportation seminar	Washington, USA	17-19 Apr.1984
54	Specialists'meeting on improved utilization of water reactor fuel with special emphasis on extended burnups and plutonium recycling	Mol, Belgium	7-11 May 1984
55	A N S Annual meeting	New Orleans, USA	3-8 June 1984
56	Technical Committee meeting on inorganic ion exchangers and adsorbents for chemical processing in the nuclear fuel cycle	Vienna, Austria	12-15 June 1984
57	25 Annual meeting of the Institute of Nuclear Materials Management	Columbus, USA	15-18 July 1984
58	A N S International topical meeting on fuel reprocessing and waste management	Jackson Hole, USA	26-29 Aug.1984
59	International workshop on irradiated fuel storage operating	Toronto, Canada	17-18 Oct.1984
60	OECD/NEA Seminar on remote handling equipment for nuclear fuel cycle facilities	Harwell, UK	22-25 Oct. 1985
61	12 Water reactor safety research information meeting	Gaithersburg, USA	23-26 Oct.1984
62	Joint meeting of the American Nuclear Society and Atomic Industrial Forum	Washington, USA	11-16 Nov.1984
63	Atomic Industrial Forum fuel cycle conference	New Orleans, USA	12-15 March 1985

No.	Title	Place	Date
64	Engineering desing of plants and processes meeting	London, UK	20 March 1985
65	Waste management - 85	Tucson, USA	24-28 March 1985
66	National Association of corrosion engineers - Annual meeting and materials performance and corrosion show	Boston, USA	25-29 March 1985
67	Executive conference on remote operations and robotics in the nuclear industry	Pine Mountain, USA	21-24 April 1985
68	CMFA VI Symposium on 'Investigations in the Area of Repro-cessing of Spent Fuel and Conditioning of Radwastes	Piestany, CSSR	22-25 April 1985
69	Physical protection technology workshop	Albuquerque, USA	23-25 April 1985
70	19 Annual aerospace mechanism symposium	San Francisco	2-3 May 1985
71	Engineering developments in reactor refuelling conference	London, UK	May 1985
72	7 Symposium on safeguards and nuclear material management	Liège, Belgium	21-23 May 1985
73	9 Annual actinide separations workshop	Richland, USA	22-24 May 1985
74	A N S Annual meeting	Boston, USA	9-14 June 1985
75	International Precious Metals Institute Conference and exhibit	New York, USA	10-13 June 1985
76	Workshop on requirements of mobile teleoperators for for radiological emergency response and recovery	Dallas, USA	23-25 June 1985
77	IAEA Technical Committee meeting on methods used in design of spent fuel storage facilities	Espoo, Finland	30 Sept.-3 Oct.85

No.	Title	Place	Date
78	Fuel Cycle Conference 1986	Sheraton, Scottsdate, Phoenix, USA	1-4 April 1986
79	Third International Spent Fuel Storage Technology Symposium/Workshop	Seattle, Washington U S A	8 - 10 April 1986
80	Technical Committee meeting on Behaviour of Used Fuel Assemblies and Storage Equipment at Long-Term Wet Storage Conditions	Leningrad, USSR	26 - 30 May 1986
81	Coordination Research Meeting on Behaviour of Spent Fuel Assemblies During Extended Storage (BEFAST)	Leningrad, USSR	26 - 30 May 1986
82	E N C' 86 Nuclear: Energy of Today and Tomorrow	Palexpo, Geneva, Switzerland	1 - 6 June 1986
83	IAEA Technical Committee Meeting on "Materials Reliability in the Back-End of the Nuclear Fuel Cycle"	Vienna, Austria	2 - 5 Sept. 1986

INIS — A GENERAL INTRODUCTION

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What is INIS?

The acronym INIS stands for International Nuclear Information System. INIS was planned and has been operated since 1970 by the International Atomic Energy Agency (IAEA) in collaboration with its Member States and co-operating international organizations in order to fulfil the IAEA's statutory obligation to foster the exchange of nuclear information among its members. At present there are 74 Member States of the IAEA and 14 international organizations participating in INIS. The System itself is a decentralized international bibliographic database providing a comprehensive nuclear information announcement and abstracting service in every aspect of the peaceful uses of nuclear science and technology and related fields.

The INIS Philosophy

The basis of INIS is international co-operation. It is the first international information system in which both the collection of input and the dissemination of output are completely decentralized. Only the processing, checking and merging of the data are centralized. This decentralized approach to input and output was selected because it should potentially:

1. result in the most comprehensive coverage of the nuclear literature;
2. provide the most effective method of handling information in different languages;
3. spread the cost of data gathering and processing equitably between large and small producers and users of the literature;
4. assist in improving the national information infrastructures in both developed and developing countries;
5. result in the most satisfactory services for users of the information.

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The INIS Liaison Officers

The INIS Liaison Officers play a key role: they are responsible for organizing the collection of information and the preparation of input on a national level. They are also responsible for setting up and maintaining national information services using the INIS products and for encouraging their utilization.

In addition, the INIS Liaison Officers provide the INIS Secretariat with advice on matters relating to the administration, operation, and development of INIS. Regular communication takes place by correspondence through the INIS Circular Letters, INIS Technical Notes and INIS Information Letters, and through the annual Consultative Meetings of INIS Liaison Officers. An important contributing factor to the success of INIS has been the spirit of mutual understanding and co-operation that has developed among individual Liaison Officers, the Liaison Officers as a group, and the INIS staff at the IAEA. A quarterly INIS Newsletter also assists in maintaining communication between the Secretariat and the Liaison Officers.

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High energy physics
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Chemistry
Materials
Earth sciences
All effects and various aspects of external radiation in biology
Radioisotope effects and kinetics
Applied life sciences
Health, radiation protection and environment
Radiology and nuclear medicine
Isotope and radiation sources
Isotope and radiation applications
Engineering
Fission reactors (general)
Specific fission reactor types and their associated plants
Instrumentation
Waste management
Economics and sociology
Nuclear law
Nuclear documentation
Safeguards and inspection
Mathematical methods and computer codes
Miscellaneous

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The INIS Document

Bibliographic databases like INIS report references to documents in the subject scope of the database, and not the document itself. Still, for simplicity the term 'document' for an entity of the database is mostly used instead of the more precise 'document reference'.

Normally the INIS document for a piece of literature contains:

1. a bibliographic description identifying authorship, publisher and similar details;
2. an abstract summarizing the information contained in the piece of literature;
3. a set of descriptors identifying the subject content of the piece of literature, selected from the "INIS: Thesaurus".

All this is done following precise standards and rules which assure the consistency of the information reported and are laid down in a series of manuals - the INIS Reference Series. Examples are IAEA-INIS-1, "INIS: Descriptive Cataloguing Rules", IAEA-INIS-3, "INIS: Subject Categories and Scope Descriptions", IAEA-INIS-12, "INIS: Manual for Indexing", and IAEA-INIS-13, "INIS: Thesaurus".

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Each issue consists of a main entry section and a number of indexes. The main entries are arranged by subject categories to permit users to scan quickly through the sections that are relevant to their subject interests in order to locate new information in their fields. Within a category, report literature is followed by journals and books. In the first issue of each volume, beginning with Vol. 12 (1980), there appears a list of the various journals scanned regularly by national INIS centres in connection with their collecting national input.

The following indexes are provided in each issue:

1. personal author index;
2. corporate entry index;
3. subject index;
4. conference index (by date and place);
5. report, standard and patent number index.

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3. INIS non-conventional literature on microfiche

Literature reported to INIS may be divided into two categories, conventional and non-conventional. Conventional literature is that literature which is commercially available through the normal distribution channels, such as the book and magazine trade or publishing houses. Non-conventional literature comprises all other forms of literature, including scientific and technical reports, patent documents, non-commercially published theses and dissertations, and standards.

The INIS Clearinghouse, a unit with the INIS Section of the IAEA, supplies on request microfiche copies of most of the non-conventional literature announced on the INIS magnetic tapes or in "INIS Atomindex" (approximately 30% of all items reported to the System).

4. The INIS online service of the IAEA

The INIS online service, like the magnetic tape service, is available only to Member States and international organizations participating in INIS. (Note, however, that subscription to "INIS Atomindex" and to the INIS microfiche service is available without limitations.) The basic components of the online service are:

1. connect-time for interactive searching of bibliographic databases;
2. automatic execution of stored search profiles, i.e. a selective dissemination of information (SDI) service;
3. automatic mailing of the printed results of searching and SDI execution.

Organizations in Member States wishing to utilize the online service must first request authorization from their national INIS Liaison Officer. Liaison Officers in many cases can also offer advice on the technical mode of access most suitable for their part of the world.

IAEA PROGRAM FOR HANDLING, PROCESSING AND STORAGE OF WASTES

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INTRODUCTION

The International Atomic Energy Agency (IAEA) has an active program in the field of radioactive waste management which is administrated by the Agency's Waste Management Section within the Division of Nuclear Fuel Cycle. The primary mission of the Agency's waste management program is the fostering of the exchange of technical, scientific and regulatory information on the subject to its 112 Member States, considering the needs of both developed and developing nations. International cooperation and exchange of information are performed through Agency-sponsored scientific conferences, technical meetings, research coordination programs and technical assistance programs.

The Agency's waste management program is organized into four subprograms covering:

- handling, treatment, conditioning and storage of radioactive waste;
- decontamination and decommissioning of nuclear facilities;
- underground disposal of radioactive wastes and management of wastes from uranium mining and milling; and
- environmental aspects of radioactive releases.

As indicated by the title of this paper, I will cover only the activities of the subprogram on handling, treatment, conditioning and storage of wastes focusing on technical meetings held in 1985 and planned for the 1986-87 period. I will also provide the latest information on coordinated research program (CRP) activities under this subprogram, discussing the scope of those programs, participation in the programs by various Member States of the Agency and the end products expected.

SUBPROGRAM ON HANDLING, TREATMENT,
CONDITIONING AND STORAGE OF RADIOACTIVE WASTE

Background

The Agency's subprogram for handling, treatment, conditioning and storage of radioactive waste covers three main elements, namely, low-intermediate-level wastes including wastes from nuclear power plants, high-level/alpha-bearing wastes and gaseous wastes. Activities in these elements have centered on the minimization of the volume of waste quantities generated, collection and control of waste streams at their point of origin, treatment of wastes to reduce their volume or removal of radionuclides and conditioning of the wastes to stable forms and packages. The principal effort of the Agency is the review, collection and dissemination of information through technical publications. Table I provides a summary of the number of documents produced since 1980 from which a distribution profile of the effort placed in each element within the subprogram can be obtained.

TABLE I

IAEA PUBLICATIONS ON HANDLING, TREATMENT,
CONDITIONING AND STORAGE OF WASTES FROM 1980

AREA	TYPE OF DOCUMENT				
	TRS	TECDOC	SS	SP/SR	TOTAL
Low-Intermediate-Level Wastes (LILW), Wastes from Nuclear Power Plants	6	3	1	1	11
Alpha/High-Level Wastes	4	1	-	1	6
Gaseous Waste Management	5	1	-	2	8
TOTAL	15	5	1	4	25

TRS = Technical Reports Series
TECDOC = Unpriced Technical Document
SS = Safety Series
SP/SR = Symposium Proceedings/Special Reports

The distribution among the subprogram elements is reflected below:

LILW, wastes from nuclear power plants - 60%
Gaseous waste management - 32%
Alpha/HLW - 24%

More than half of the activities of the Agency in this subprogram have been placed on low-intermediate- level wastes and wastes from nuclear power plants reflecting the high interest placed by Member States on the management of waste from nuclear power plants. Of course, future distribution of effort may change depending on the interests of the Member States.

I would now like to cover each element of the Agency's subprogram on handling, treatment, conditioning and storage of waste by providing a review of recent meetings held in 1985 and planned meetings for 1986 and 1987.

Low-Intermediate-Level (LIL) Wastes and Wastes from Nuclear Power Plants

With the high interest in the management of low- intermediate-level wastes, especially with Member States pursuing nuclear power programs, an extensive program has been carried out in this area. Table II lists the Agency's activities as accomplished and planned.

Without going into details of every meeting, I should like to point out that, on the average, each of the above technical meetings is attended by approximately 15-20 technical experts representing Member States of the Agency and other international organizations with an interest in radioactive waste management. The end product of the meeting is usually a technical publication which presents the latest "state of the art" information in the field or a Safety Series guide providing recommendations and/or guidelines of a regulatory nature. For example, the meetings planned on the design of radioactive waste management systems at nuclear power plants will result in the publication of an Agency Safety Series document in 1987 which will provide basic criteria, guidance and requirements for Member States on the design of waste management systems at nuclear power plants.

Another meeting listed in Table II that I should like to comment on is the meeting scheduled for 1986 on management options for low- intermediate-level wastes. This meeting is planned as a regional seminar to be held in Rio de Janeiro, Brazil, in October 1986 and is classified as a teaching seminar designed especially for the developing nations in the Latin America region.

TABLE II

ACTIVITIES FOR LIL WASTES/WASTES FROM NPPs PROGRAM

Meetings Held in 1985

- Design of radioactive waste management systems at NPPs
- Handling and treatment of radioactive waste from unplanned events at NPPs

Meetings Held and Scheduled for 1986/1987

- Design and operation of radioactive waste incineration facilities
- Handling and treatment of radioactive waste from unplanned events at NPPs
- Management options for LIL wastes
- Immobilization of LIL waste with polymers
- Solidification of organic radioactive waste
- Design and operations of cementation systems for conditioning of radioactive wastes.

The Agency has twenty Member States in this region involved in varying stages of nuclear energy with different waste management problems. This seminar is intended to provide information and promote the exchange of experience gained in the management of low- and intermediate-level radioactive wastes generated from both non-fuel cycle and fuel cycle activities. The seminar will cover the activities involved in handling, treatment of wastes to reduce volume, conditioning for storage and/or disposal, transportation and disposal. Special emphasis will be placed on selection and combination of the technology options available into an integrated waste management program.

Looking beyond 1987, we are aware that close to 400 GW(e) will be generated from nuclear power by the year 1990. Included in this total is the nuclear power capacity of 15 developing nations that will be operating nuclear power plants by the end of this decade. A major effort of the Agency will likely be the dissemination of the latest information on the treatment and conditioning of nuclear power plant wastes to Member States with emerging nuclear power programs.

Alpha-bearing and High-Level Wastes

The Agency's activities in alpha-bearing and high-level waste management have been guided by the interests of Member States with mature nuclear power programs. Efforts have focused on alpha-bearing waste treatment and conditioning, handling and storage of high-level liquid wastes, solidification of high-level wastes and the characterization of solidified high-level waste products. In 1986, the Agency plans to hold an Advisory Group meeting on the treatment of alpha-bearing wastes. Selected experts from several Member States will be invited to Vienna to discuss the latest technology in this field and to recommend to the Agency the need for and direction of further work in this area.

The Agency plans to place additional emphasis on the management of alpha-bearing wastes and high-level wastes by sponsoring several technical meetings in this area in the post-1986 period. Treatment and conditioning technologies for alpha-bearing wastes, technology and environmental factors in comparing spent fuel vs. immobilized waste as a final HLW form, and techniques, testing methods and methodologies for the evaluation of conditioned HLW forms are examples of technical meetings planned in this area.

Gaseous Waste Management

Now, turning to gaseous waste management, this area has been an active part of the Agency's program. Technical meetings and publications of documents on the technologies for retention of iodine and other airborne radionuclides and methods and equipment for testing off-gas and exhaust air cleaning systems have been a major part of the past program. In 1985, a shift in program emphasis began with technical meetings on the management of gaseous waste at nuclear fuel cycle and waste treatment facilities. Table III lists the meetings held in 1985 and those planned for 1986 and 1987.

TABLE III

ACTIVITIES FOR GASEOUS WASTE MANAGEMENT

Meetings Held in 1985

- o Conditioning, storage and disposal of Iodine-129
- o Design and operation of off-gas cleaning systems at waste conditioning facilities

Meetings Scheduled for 1986/1987

- o Design and operation of off-gas cleaning systems at LILW treatment and conditioning facilities
- o Design of ventilation and air cleaning systems at non-fuel cycle facilities

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Emphasis on gaseous waste technology in nuclear fuel cycle facilities, other than nuclear power plants, will continue in 1986 and 1987. Three technical meetings are planned relating to gaseous waste management at fuel reprocessing and waste conditioning facilities.

Looking beyond 1987, we expect that the major activity will focus on developments in gas and air cleaning systems in nuclear power plants as the state of the art continually improves. New developments in testing and monitoring gas cleaning systems in nuclear facilities will also be observed closely and a technical meeting or seminar will be scheduled to coincide with technology advances in this field.

Coordinated Research Programs

Up to this point, I have been discussing the Agency's use of technical meetings to disseminate information to its Member States. As mentioned in the Introduction to this paper, the Agency also sponsors Coordinated Research Programs which have the prime objective of fostering cooperation and the exchange of R/D information among Member States that have a common interest in a particular subject. Coordinated Research Programs are normally organized by the Agency with Member States through the execution of research contracts or research agreements on a subject of common interest. Research contracts are provided to developing Member States and differ from research agreements, which are offered to developed Member States, in that a small part of the R/D cost is supported by the Agency in the former case. Coordinated Research Programs usually have from 5 to 10 Member State institutes participating and normally extend over a 4 or 5 year period. Three times during the course of the program, the principal investigators under the program are invited to a research coordination meeting at the Agency's expense. At these meetings, the program of each participant is discussed in terms of content, direction and achievements. Our experience with CRPs shows them to be an extremely effective mechanism for keeping interested Member States well informed in an area of particular interest as well as providing the opportunity for them to receive valuable feedback on their own program efforts from experts of other countries. Under the Agency program for handling, treatment, conditioning and storage of radioactive waste, three Coordinated Research Programs are active or in the final planning stages (Figure 1).

On-going or Planned CRPs	MEMBER STATES INVOLVED	1984	1985	1986	1987	1988	1989	1990
1. Retention of iodine and other airborne radionuclides during abnormal and accident conditions	Austria, Belgium, FRG, GDR, Hungary, India, Republic of Korea, Yugoslavia	RCM		RCM	RCM	P		
2. Performance of solidified HLW forms and engineered barriers under repository conditions	Australia, Belgium, Canada, China, FRG, Japan, India, Sweden, UK, USA	PE	RCM		RCM	RCM	P	
3. Evaluation of low- and intermediate-level solid waste forms and packages	12 Member States		PE	RCM		RCM	RCM	P

KEY PE = Programme Established
RCM = Research Coordination Meeting
P = Publication

Fig. 1 Coordinated Research Programs.

Retention of gaseous radionuclides during abnormal and accident conditions has raised considerable interest in the international nuclear community. Questions on the behavior of filters and other off-gas cleaning systems under accident conditions have fostered new work in this area. To support this effort and provide for the effective exchange of information among interested Member States, the Agency initiated a CRP in 1983 on the "Retention of Iodine and Other Airborne Radionuclides during Abnormal and Accident Conditions".

This CRP is composed of eight Member States and is expected to be active through 1987. The first research coordination meeting under this program was held in Mol, Belgium in September 1984. Work on the testing of off-gas systems during abnormal conditions, including the retention properties of various absorbent materials as affected by age, humidity, face velocity, etc., will be covered during the course of the program. The second meeting is planned to take place in Ontario, Canada, in June of this year.

The CRP on the "Performance of Solidified HLW Forms and Engineered Barriers Under Repository Conditions" has recently been established and is a follow-on program to a recently completed CRP entitled "Evaluation of Solidified High-Level Waste Forms". The first CRP which was completed in 1983 had the primary objectives to review and disseminate information on properties of solidified high-level waste forms, to provide a mechanism for analysis and comparison of results from different institutes and to establish future plans for work in this area. The results of the CRP are contained in an Agency Technical Reports Series document which was available for distribution by the end of 1985. The document titled "Chemical Durability and Related Properties of Solidified High-Level Waste Forms" presents an up to date summary of the state-of-the-art in this field. As shown in Figure 2, the new CRP on the performance of solidified HLW forms has now been established. I am pleased to report that eleven institutions from ten Member States are participating in this program. The first research coordination meeting was held in Tokyo, Japan, in late October 1985 where initial results of work were presented and plans developed for follow-up meetings. To obtain some understanding of the scope and content of this CRP, the following list presents a few examples of the work planned by the participating institutions:

- properties and performance of HLW glass products and engineered barriers;
- performance of SYNROC under conditions relevant to repository disposal; and
- performance of conditioned spent fuel under repository-relevant conditions.

An interesting point is to recognize that the scope of the CRP covers the evaluation of conditioned spent fuel as well as solidified HLW forms.

The Coordinated Research Program entitled "Evaluation of Low- and Intermediate-Level Solid Waste Forms and Packages" has now been formulated.

This CRP was established last year and is composed of institutions from Member States. The objective of this program is to exchange information and promote new work on the solidification of low- and intermediate-level waste and the evaluation of the resulting waste forms and packages. Types of wastes in the program scope include spent ion-exchange resins, concentrates and sludges from nuclear power plants or other nuclear facilities. The first research coordination meeting under this CRP will be held in Egypt in the middle of this year.

CONCLUDING REMARKS

This paper on the IAEA's program on the handling, treatment, conditioning and storage of radioactive waste provides a brief overview of current and planned activities by the Agency in this field. This program is being implemented by the Agency through the contributions and continued support of the Member States and in close cooperation with other international organizations involved in radioactive waste management. The Agency's efforts to maintain an effective international forum for the exchange and dissemination of information on the management of radioactive wastes depend on its receiving the necessary input from Member States as to how best to employ its resources.

TRANSPORTATION OF SPENT FUEL

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1.0 INTRODUCTION

It is currently estimated that between 20 million and 40 million shipments of packages of radioactive material are made each year worldwide; and only a very small percentage of these are of spent fuel. Historically, the safety record for the shipments of radioactive material in general and of spent fuel specifically is exceptional; primarily as a result of the regulations used by individual countries and international organizations to control the packaging, handling, storage and shipment of such consignments.

Shortly after the formation of the International Atomic Energy Agency (IAEA) in 1957, it was given the task of developing safety rules for the transport of radioactive materials, covering all modes of transport. As a result, and with the assistance of experts worldwide, the first edition of the IAEA's Regulations for the Safe Transport of Radioactive Materials was published as Safety Series No. 6 in 1961. Revised editions of Safety Series No. 6, which took into account developments in technology and shipping practices, were issued in 1964, 1967, 1973 (with an amended 1973 edition being also issued in 1979), and most recently in 1985. Safety Series No. 6 is used as a model (i.e., it serves as recommendations) for domestic transport regulations of countries throughout the world, and it also serves as a model for the regulatory documents issued by various international transport organizations such as the International Civil Aviation Organization (ICAO) and the International Maritime Organization (IMO).

2.0 CHANGES IN THE 1985 EDITION IMPACTING SPENT FUEL TRANSPORT

Three major changes could affect the transport of spent fuel in the future, as follows:

(a) Release requirement on Type B(U) package -- the release requirement (following exposure to the accident-simulating conditions) was made less restrictive [from 0.001 A_2/wk to A_2/wk] so the requirement for a Type B(U) package is now equivalent to the requirement for a Type B(M) package.

(b) Deletion of Fissile Classes -- Nuclear criticality control for fissile materials has been coupled to the Transport Index and therefore is inherently now a part of the package Category descriptor; this action greatly simplifies the operational controls and better assures compliance and safety.

(c) Adoption of a 200 m water immersion test for irradiated fuel casks -- a requirement for a 200 m water immersion test for packages containing irradiated fuel having more than 37 PBq (1 MCi) activity; the requirement is that following immersion there will be no rupture of the containment system which will facilitate recovery of the cask should it be lost in relatively shallow water (e.g., in rivers, most lakes, along sea coasts and on continental shelves).

3.0 PLANS FOR ADOPTION OF THE 1985 EDITION OF SAFETY SERIES NO. 6

Recently the IAEA requested information from its Member States on how they regulate the transport of radioactive material and what their plans were for the adoption of the 1985 Edition of Safety Series No. 6.

Briefly, it was determined that approximately 15 percent of the Agency's Member States regulate transport using only the IAEA's Regulations, 20 percent regulate using only the regulatory documents of other international organizations (of which Safety Series No. 6 is the basis) and 65 percent regulate using both. The key international organizations (ICAO for air, IMO for sea, OCTI/RID for road and ECE/ADR for rail) all plan to implement the 1985 Edition of Safety Series No. 6 on 1 January 1990. Data from the above-mentioned survey indicate that approximately 85 percent of the Member States responding (42 in all) will have adopted or otherwise implemented the 1985 Edition of Safety Series No. 6 by 1990.

4.0 FUTURE REVISIONS OF SAFETY SERIES NO. 6

Beginning in 1986, the IAEA will issue a supplement to Safety Series No. 6 every two years which will provide minor changes and changes of detail thereto. The next major revision of Safety Series No. 6 is not anticipated until 1995 or later.

5.0 SPENT FUEL SHIPMENTS

Full data are not available to the Agency on the shipment of spent fuel worldwide. Suffice it to say that there are many countries, organizations and companies involved in the design, development and operation of spent fuel casks, and in the loading, shipping and unloading of spent fuel.

Two examples of shipping data are provided:

(a) Spent fuel shipping in the USA is briefly summarized (Reference 1) as follows:

   (i) Since 1964, average number of shipments has been 291 shipments per year;

   (ii) The average number of shipments in recent years has been much lower than this;

   (iii) There have been approximately 5000 shipments of spent fuel in the USA to date.

   (iv) When a repository becomes operational in the USA (in about the year 2000?), the maximum number of shipments could reach 9000 per year, with an average of about 4500 per year.

(b) Spent fuel shippping within and into France is summarized (Reference 2) as follows:

   (i) Shipments to LaHague and Marcoule are very large and the only other facility receiving comparable amounts would be Sellafield;

   (ii) Through 1984, more than 10,200 tonnes of spent fuel were received at the French facilities;

(iii) Shipments in 1984 were as follows:

approximately 500 tonnes of graphite gas reactor fuel, and

approximately 800 tonnes of LWR fuel.

(iv) LWR shipments to the French facilities is expected to stabilize at approximately 1200 to 1500 tonnes per year, i.e., about 250 shipments per year.

6.0 REFERENCES

1. R.M. Jefferson, "Transporting Spent Fuel - Considerations for Safety", IAEA Bulletin, Spring 1985.

2. B. Lenail, "Update of French Experience Shipping Irradiated Fuel", IAEA Bulletin, Spring 1985.

THE NEW INITIATIVES FOR INTERNATIONAL CO-OPERATION IN THE FIELD OF SPENT FUEL MANAGEMENT: WHICH PROJECTS AND WHY?

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In the nearest future (1987-1990) IAEA plans to initiate a number of new projects in the framework of Spent Fuel Management (SFM) programme. The needs for modification of "traditional" directions of activity are dictated by the following factors:

- strengthening of SFM impact on the development of atomic industry as a whole and increasing of their sensitivity with a view of public acceptance;
- identificaton of a number of new problems (or, more exactly: the old problems in the light of modern knowledges) on the basis of both accumulated world experience and future tendencies;
- objective necessity to increase technical assistance to those countries which only intend to start nuclear programmes in order to avoid former mistakes.

Based on the analysis of the present situation and some future trends in the world spent fuel storage, transportation and treatment technology (including recycling of fissile materials), and in accordance with a real possibility of the Agency as an international coordinator, a number of topics were determined (as a subject for discussion), which, in principle, could have sufficient influence on the harmonization of this important part of the Nuclear Fuel Cycle.

I. Spent Fuel Treatment

During the last years the Agency's activity in this area was very limited having in mind the sensitiviy of the reprocessing technology with a view of non-proliferation policy. In general, the situation as regards the technology of Pu recovery has not changed at present. But there is at least one subject which is of great importance to the "nuclear" world and, on the other side, cannot directly influence the non-proliferation strategy.

It is well known that used nuclear fuel is a unique source of valuable products for modern technology. In particular, irradiated fuel elements, in contrast to common belief, does not contain wholly radioactive materials: the vast bulk of the material is not radioactive and is contaminated with a small amount of highly radioactive isotopes only. Of essentially non-radioactive fission products, the most attractive from the economic point of view are the platinum group metals - rhodium, palladium and ruthenium (table 1). All three elements exhibit distinctive properties (Annex A) and are frequently used in the chemical,

Table 1. Range of prices for platinum-group metals (in dollars per gram)

Element \ Year	1978	1979	1980	1981	1982	1983	1984	1985	Average 1978-85	Average 1958-68	% change
Ruthenium	2,0[a]	1,5[a]	1,5[a]	1,5[a]	1,5[a]	1,5[a] 0,9[b] 1,6[c]	1,5[a] 3,4[b] 5,14[c]	5,4[a]	2,3	1,9 [d]	+ 21
Rhodium	16,1[a]	25,7[a]	25,7[a]	19,3[a]	19,3[a]	19,3[a] 10,1[b] 11,0[c]	19,3[a] 19,8[b] 29,2[c]	19,3[a] 31,2[c]	20,4	5,3 [d]	+ 384
Palladium	2,3[a]	3,4[a]	7,2[a]	4,2[a]	4,2[a]	4,3[a] 4,4[b]	4,8[a] 4,8[b] 4,7[c]	4,8[a] 3,9[c]	4,4	0,9 [d]	+ 488

a - ref 1, b - ref 2, c ref 3, d - ref.4

Table 2. Typical computed data for the concentration of "artificial platinum-group metals in spent fuel after 10 years storage (g/t spent fuel)

Reactor system, source / Element	L W R 33.000 $MW \cdot d \cdot t^{-1}$					L M F B R 100.000 $MW \cdot d \cdot t^{-1}$		
	IAEA	CMEA	USA-I	USA-II*	Average	IAEA	CMEA	Average
Ruthenium	2050	2140	2170	2270	2158	2680	2660	2670
Rhodium	320	392	470	470	413	700	874	787
Palladium	1140	1410	1380	1240	1292	2040	2090	2065
Total PGM					3863			5522

IAEA-ref.5; CMEA-ref.6; USA-I-ref.7; USA-II-ref 1

* Results obtained for burnup = 30.000 $MW \cdot d \cdot t^{-1}$

automobile, petrolium, glass and electrical industry, in the dental and medical field, etc. (Annex B). The world natural reserves of these elements are very limited: e.g. the rhodium reserves are only about 770 tonnes, and the concentration of rhodium in the earth's crust is approximately 1.10^{-7}% (1.10^{-3}g/t). At the same time, concentration of platinum group metals in used fuel elements is almost 4 kg/t (tables 2 and 3).

Table 3. Composition of "Artificial" Platinium-Group Isotopes in Spent Nuclear Fuel /1/

Element	Isotopes-Fission Product	Half - life	Composition (wt%) in Spent Fuel after Discharge	
			1 year	30 years
Ruthenium	99	stable	$2,4\times10^{-4}$	$3,6\times10^{-3}$
	100	stable	4,2	4,37
	101	stable	34,1	35,42
	102	stable	34,0	
	103	39,4 days	$3,6\times10^{-3}$	
	104	stable	23,9	24,9
	106	368 days	3,8	$8,5\times10^{-11}$
Rhodium	103	stable	100	100
	103m		$1,6\times10^{-5}$	
	102m	3 years	trace	
	106	2,18 hours	$1,8\times10^{-6}$	
Palladium	104	stable	16,9	15,8
	105	stable	29,3	27,4
	106	stable	21,3	26,4
	107	$6,5\times10^{6}$ years (0,035 MeV)	17,0	15,9
	108	stable	11,7	10,9
	110	stable	3,8	3,6

The net annual availability of only rhodium and only from the US power reactor fuels are more than 2000 kg/year. Taking into account that by the year 2000 more than 200 000 tonnes of used fuel all over the world will be discharged, the "new artificial" reserves of rhodium will reach 68 000 kg. Almost the same situation is with palladium and ruthenium. In other words, in the 21st century, used nuclear fuel may become the unique source of these vital elements (!). Here may be mentioned also such important elements for world industry and medicine as technetium, cesium-137, strontium-90 and promethium.

The main reasons for starting an international activity in the field of recovery and utilization of valuable non-fissile elements from used nuclear fuel are the following:

- Obvious possiblity and necessity to provide humanity with vital resources in the future;
- Necessity to intensify the nuclear application development on the basis of rational use of a large amount of radioactive isotopes;
- Potential danger that exisitng spent fuel and waste management strategy will make it impossible to use tremendous and still unknown reserves for industrial progress in the future.

Thus, international "exploration" of these articificial reserves is needed right now, and it might be even too late already. It does not mean that the Agency intend to initiate R & D on the special used fuel treatment technology with emphasis on recovery of non-fissile elements only; it does not mean that it is necessary to change immediately the national approaches to the back-end strategy. The main aim of the new project is to create a scientific basis for the future technical solutions on this, undoubtedly, important problem, and to evaluate a real potential of valuable sub-products in used fuel and real industrial demand of these elements. Therefore, in 1987-88 the Agency plan to start (on the basis of consultations) the evaluation of noble metals, Tc, Cs and Sz arisings and demand.

II. Recycling of fissile materials

The economic advantages of light water reactors (LWR) - the most widespread type of nuclear reactors for the time being - are undoubtful, but their obvious drawback is that they use only a small proportion of the total energy potential in uranium fuel: they only achieve a fuel utilization factor of about 0,6% at a conversion ratio of 0.6 and, consequently, have a natural uranium consumption of 4220 tonnes of U (natural) over 30 years of operation at an average load factor of 70%. This apparent handicap is not, however, fundamental, since the spent fuel can be regenerated by appropriate treatment, separating the fission products from residual uranium and from a new fissile material (Pu) generated during the irradiation process. The regenerated products can then be recycled either in the same LWR (Fig.1), or in the fast breeder reactors (FBR).

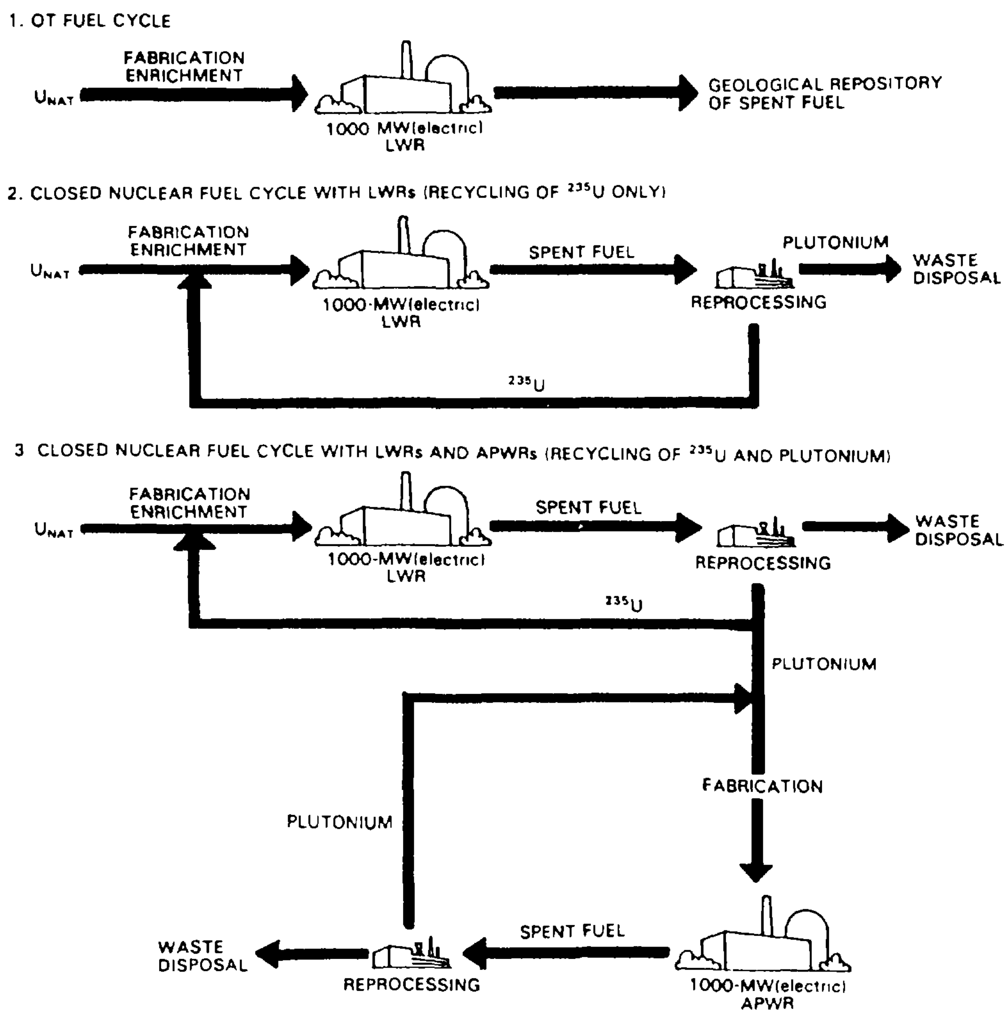

Fig. 1. Alternative fuel cycle strategies. /8/

At present the uranium situation is characterised by near-term over-supply and low prices. But only in WOCA countries the demand in uranium would increase from about 31.000 tonnes in 1984 to 47.000 tonnes in 1990 and 57.000 tonnes in 1995. As the lead times for uranium mining projects are very long, over ten years, higher levels of uranium exploration will be needed if this demand is to be met. Moreover, looking in the future - the 21st century - one may ask, whether the nuclear community could guarantee the same situation after 20-30 years, or maybe already now it is time to intensify cooperative efforts in the field of uranium and plutonium recycling in order to provide a stable development of nuclear power. The answers may be manifold but it is necessary to keep in mind the objective fact, that more and more countries are crossing from theoretical investigations to industrial introduction of recycling technology.

IAEA was involved in this activity since 1964. In particular, the Agency has convened an international Symposium on plutonium utilization in thermal power reactors in 1967 and four panels (1964, 1968, 1971 and 1974) on this subject. Taking into account the importance of this subject (the new development stage in the field of MOX fuel) and potential impact of the use of recycled materials on industrial scale on all links of the nuclear fuel cycle, including SFM, it seems reasonable to overview in 1988-1990 on an international level the "Status of MOX fuel utilization in LWRs" and "Remote technology of fuel fabrication with use of fissile recycled materials". Both documents, which could be produced with assistance of specialists from countries with the most advanced technology in this field, will have an obvious aim to define the main results achieved, problems and prospects in this field that will allow to understand which kind of international cooperation will be needed and to formulate actual themes for such cooperation. Besides, for developing countries the planned reports will be very useful for the evaluation of modern experience and trends in nuclear technology and for revision of their own SFM strategies.

III. Reliability of structural materials in the Back-end of the Nuclear Fuel Cycle

The problem of materials reliability always was, is and will be one of the most sensitive subjects in any branch of techniques, and especially in nuclear technology. The different aspects of this important problem are constantly under the intense attention of SFM specialists, and usually are the main topics of international cooperation. It is enough to mention the very productive Coordinated Research Programme "Behaviour of Spent Fuel under Extended Storage", which is in principal oriented on the reliability of spent fuel claddings and equipment of spent fuel storage facilities; the planned IAEA Technical Committee Meeting on Materials Reliability in the Back-end of the Nuclear Fuel cycle (September 1986), etc.

While analysing the upcoming requirements for structural materials in spent fuel technology, it is necessary to remember that in future radiation effects will play a more and more important role. The reason for this conclusion is, first of all, the objective fact, that such innovations as increasing of the fuel burn-up, compacted storage of spent

fuel (including "rod consolidation" concept), use of regenerated uranium and plutonium for MOX fuel fabrication, etc. will result in the sufficient increase of the dose rate. It is a well known fact that such heterogenious processes as corrosion of metals and alloys, solution of metal oxides and dielectric materials in water media, synthesis of the new substances on the interphase boundaries, surface segregation, catalitic decomposition of some typical components of structural materials are promoted by ionizing radiation many times more intensively than the physico-chemical reactions in the bulk./9 - 12/
For example, the rate of the so-called "atmospheric" corrosion can increase under irradiation at 10-100 times (!).

Undoubtedly, the nature of these phenomena and the methods for prevention of such intensive destructive processes may and should be investigated carefully during the elaboration of technical solutions on spent fuel storage technology (especially for dry storage concepts); spent fuel treatment; radioactive waste treatment and disposal technology and technology of fuel fabrication from high-radioactive recycled materials. In other words, radiation induced heterogenious processes are a specific but important topic (with a view of materials reliability in the back-end of the Nuclear Fuel Cycle) for close international cooperation, which can be started already in 1987 within the framework of the coordinated research programme on "Behaviour of structural materials under irradiation with emphasis on heterogenious processes".

Apparently, one of the fruitful ways is to unite in terms of one CRP joint efforts on research both metals and non-metalic materials (insulators, semi-conductors), which have now or will have in future practical applications in nuclear technology. This approach will allow to comprehend a wide spectrum of substances with various electronic and crystallographic structures, various chemical natures and various surface properties so that it will be possible to create a general model of radiation effects in non-homogenious systems or, at least, to obtain the initial datas for such a model. Another way is to focus the attention only on one class of the structural materials (e.g. Fe-Cr-Ni alloys and steels). In this case one may expect that the obtained results will mostly reflect the real "technological" experience as behaviour of materials already used in existing storage, reprocessing and disposal facilities. The last methodological approach has obvious advantages because the results of national investigations may be discussed in depth in a homogenious professional auditorium. On the other hand, limitation of activity can hinder to comprehensive understanding of the fundamental nature of radiation effects in heterogenious systems and, as a consequence, it may be an obstacle in the way of elaborating the effective methods for "inhibition" of potential dangerous radiation induced degradation of structural materials.

In any case, there is an objective need in such a coordinated research programme, and there is confidence that cooperative efforts will help Member States to solve best their own problems in the back-end of the Nuclear Fuel Cycle.

Conclusion

The above subjects could be included in the IAEA Nuclear Materials and Fuel Cycle Technology programme as additional tasks without cutting the existing activity on spent fuel storage facilities (methodology, design criteria, economics, methods and technology of compacted storage, decontamination, etc.); on spent fuel arisings and behaviour (including surveillance and monitoring, investigations on cladding degradation under extended storage, harmonization of terminology etc.) and at last, on the general strategy of spent fuel management.

At a first glance the last question seems to be a minor subject for international discussions because it is assumed that there are only two objective options: reprocessing with the use of regenerated valuable materials and disposal of radioactive waste or direct disposal of spent fuel. But this is a "world strategy", as the spectrum of options for each individual country is much larger. For instance, spent fuel can be transfered abroad for reprocessing, for strorage or for disposal; regenerated fissile materials can be returned for recycling in the producer-country or used in the reprocessor-country, or stored under international control (e.g. see the last proposal of Dr.H.Blix). Due to the particular properties of spent nuclear fuel such specific technico-economical connections between selected countries have a number of important social and political consequences which may influence not only the bilateral relations but may also have great resonance all over the world. It is also very important to understand that a short-term choice in SFM is frequently in conflict with long-term objectives and needs and this problem can only be solved best by means of close internatioanl cooperation.

Thus, there is no doubt that the "global" SFM strategy is a "hot" subject for international discussions and this was confirmed once more during the IAEA Scientific Advisory Committee in December 1985. There is hope that the above proposed directions of activity have not only a specific technical importance but will also be useful for the harmonization of SFM stragegy as a whole.

ANNEX A

The main properties of Ruthenium, Rodium and Palladium

The members of the so-called platinum group (the second triad of group VIII or the transition group in the periodic table), ruthenium, rhodium and palladium, exhibit distinctive properties, including resistance to chemical attack, excellent high temperature characteristics, the unique ability to catalyse chemical reactions, and stable electrical properties (table A 1).

The unique properties of the platinum-group metals can be understood on the basis of binding and ionization energies examination. The valence electrons in these elements are shared between ns and (n-1)d orbitals; the former are higher in energy. Among these three elements, one or two electrons that might have been expected to occupy ns orbitals actually are gound in (n-1)d orbitals - lower energy, i.e. more difficult to remove. Hence, ionization energies are higher than otherwise expected because of the shift of the electrons to the lower orbitals.

The magnitude of the binding energy reflects the strength ofthe metallic bonding in these elements, and is governed by the number of valence electrons actually participating in the multicenter metallic bonding, the effective nuclear charge, and the size of the atoms. The first of these criteria proves to be of importance within the platinum-group metal triads. As the number of d electrons increases with an increase in atomic number, the number of electrons actually available for metallic bonding may decrease. The added "d" electrons go into spin-paired states in atomic orbitals and are not available to

Table A 1. The main practical properties of the Platinum-Group Metals (Ru, Rd and Pd) /14/

Characteristic/ Property	Element		
	Ruthenium	Rhodium	Palladium
1	2	3	4
atomic no.	44	45	46
atomic weight	101,07	102,91	106,4
melting point, ^{o}C	2310	1960	1552
specific gravity, $g \cdot cm^{-3}$	12,45	12,41	12,02
usual valence	3,4,6,8	3	2,4
atomic structure	$^{4}S2\ ^{4}p6\ ^{4}d7\ ^{5}S1$	$^{4}S2\ ^{4}p6\ ^{4}d8\ ^{5}S1$	$^{4}S2\ ^{4}p6\ ^{4}d10$
binding energy, eV	6,63	5,76	3,91
ionization poten., eV	7,36	7,46	8,33
"natural" isotopes (abundance, %)	96 (5,53) 98 (1,86) 99 (12,7) 100 (12,6) 101 (17,1) 102 (31,6) 104 (18,5)	103 (100)	102 (0,8) 104 (9,3) 105 (22,6) 106 (27,2) 108 (26,8) 110 (13,5)
crystal structure at (^{o}C)	hexagonal (20)	fcc (25)	fcc (20)
a, nm	0,27056	0,3803	0,389
linear coefficient of thermal expansion at $20^{o}C$ per ^{o}C	$9,1 \cdot 10^{-6}$	$8,3 \cdot 10^{-6}$	$11.1 \cdot 10^{-6}$
specific heat at $0^{o}C$, $J \cdot g^{-1}$	0,2306	0,247	0,244
thermal conductivity, W/(m·K)	-	150,6	75,3
vapor pressure at melting point, Pa	1,31	0,133	3,47
electrical resistivity, $\mu\Omega \cdot cm$ at $0^{o}C$ at 20°	6,80 7,40	4,10 4,50	9,93 9,96
temperature coefficient of electrical resistivity, 0-$100^{o}C$, per ^{o}C	0,0042	0,0046	--
mass susceptibility, $cm^{3} \cdot g^{-1}$	$0,43 \times 10^{-6}$	$0,99 \times 10^{-6}$	$5,23 \times 10^{-6}$
work function, eV	-	4,80	4,99
tensile strength annealed, MPa	4,96	758,6	165,5
Young's modulus of elasticity, GPa	413,8	344,8	117,2
elongation, %	-	30	24-30
Vickers hardness, annealed	220	120	37-39
Poisson's ratio	-	0,26	0,39
Emf vs Pt at 1000°C, mV	9,760	14,12	-11,491

participate in the multicenter band orbital characteristic of metallic bonding. This explains the falling of in binding energy with increase in atomic number within triad.

The more important properties of palladium, rhodium and ruthenium are shown in table A 1. Some details:

Palladium acquires a superficial coating of oxide when heated in air from 350 to about 790°C; above this temperature the oxide decomposes and leaves the bright metal. Palladium dissolves a small amount of oxygen at elevated temperatures, and if this type of alloy is then heated in hydrogen, water vapor is formed within the metal, disrupting the grain boundaries. The surface can also be damaged by torch flames, particularly if they are reducing flames, but this can be avoided almost completely by using acetylene as the fuel gas. Since the sulfur compounds present in urban atmospheres have no effect upon palladium, it is used for electrical contacts, especially for telephone circuits, and for jewelry. Even exposure to pure sulfur dioxide at 800 and 1000°C for one hour produces only a moderate weight gain but no embrittlement. Hydrogen sulfide at temperatures above 600°C attacks the metal, producing a low melting entectic. A property unique to palladium is its ability to absorb and retain over 800 times its volume of hydrogen, which results in an expansion of several percent. Large quantities of hydrogen can be introduced by making palladium the cathode in a dilute acid electrolyte. Such palladium is an active reducing agent that reduces palladium from its chloride solution. Palladium dissolves anodically in warm acidic chloride solutions - this property is used in maintaining the palladium content in electroforming baths, all-chloride as well as double-nitrite plus potassium chloride. Palladium is attacked by nictric acid, particularly if a trace of chloride is present, and dissolves in hot concentrated sulfuric acid at about 300°C.

Rhodium remains bright in all atmospheric exposures at ordinary temperatures and is completely resistant to a variety of corrosives. These properties plus the high and relatively uniform reflectivity (75-80%) and the ease with which hard, bright electrodeposits can be produced have led to wide use of rhodium for jewelry, search light reflectors, and electrical contacts. Thin reflective coatings are produced by sublimation. Upon heating in air, rhodium oxidizes slowly at 600°C, the rate reaching a maximum at about 800°C; above 1000°C, the oxide decomposes. The rhodium-platinum alloys containg up to 20%

rhodium are the preferred platinum-metal alloys for high temperature service.

Rhodium sulfide is more stable at high temperature than platinum sulfide. Thus, a 10 % Rh-Pt alloy is attacked in sulfur vapor at 1100°C whereas the 3,5% alloy used for crucibles not attacked. At 100°C aqua regia does not attack rhodium whereas strong sulfuric acid, hydrobromic acid, and sodium hypochloride attack slightly. Hot sulfuric acid and fused bisulfates attack rhodium at a rate sufficient to be useful in refining, and free halogens react at 200-600°C. Molten cyanide at 550°C attacks rhodium more vigorously than other platinum metals, whereas molten alkali nitrates at 350°C attack rhodium only mildly; alkali metal peroxides are more effective. The solubility of rhodium in molten lead, its solubility in hot concentrated sulfuric acid, and the stable complex chlorides and nitrites are utilized in refining and analysis. Rhodium can be reduced to the metallic form by treatment with titanous compounds or hydrazine in a hot, slightly alkaline solution. It can also be precipitated in an impure state by zine or magnesium. Its complexes with ammonia or ammonium salts are decomposed by heating, reduction with hydrogen, and cooling in an inert gas. An inert atmosphere is often necessary when cooling finely divided metals containing hydrogen to avoid self-ignition when subsequently exposed to air. Very pure rhodium is dictile when hot and less so when cold, although it hardens very rapidly on cold-working and is not comparable with platinum or palladium in workability.

Ruthenium is highly resistant to corrosive agents at moderate temperatures and is unattacked by aqua regia, sulfuric, hydrochloric, hydrofluoric, and phosphoric acids at 100°C. Potassium cyanide solution and mercuric chloride solution attack at 100°C whereas, chlorine water, bromine water, and iodine in alcohol attack only very slightly at room temperature. Many molten salts attack ruthenium less than rhodium, but sodium peroxide attacks rapidly and might be regarded as a specific corrosive. Ruthenium oxidizes slowly in air above 450°C, forming the slightly volatile dioxide. Ruthenium tetroxide is produced when alkaline ruthenium salts are treated with chlorine or bromate; this oxide is volatile and may explode. Ruthenium compounds usually are reduced to metal by heating ammonium hexachlororuthenate (IV) in hydrochloric acid plus methanol to the oxydichloride, which can be reduced in hydrogen to the metal.

ANNEX B

The Principle Uses of Platinium-Group Metals

Palladium The main consumers of palladium are the electrical (39%)*), chemical (26%) and automotive (18%) industries, dental and medical (14%), jewelry (2%) and petrolium processing (1%).

In electrical contacts, the high melting temperatures (table A 1) of palladium and palladium alloys provide high resistance to arc erosion and the welding of mating-contact surfaces. In fact, in many electrical and electronic applications, palladium or palladium alloys are considered a likely substitute for gold, depending on the prevailing prices.

For dental and medical applications, palladium is used in gold-based alloys. The palladium content varies from 2% in yellow gold to 40% in white gold. Palladium-rich alloys also are used as support in porcelain-overlay bridgework. The use of palladium load bearing devices for replacement of damaged bones and joints is expected to increase significantly.

The automotive industry has used palladium in catalitic converters since 1974. Palladium catalysts are employed in hydrogenation or dehydrogenation reactions supported on activated carbon. The carbon is removed easily by burning when the catalyst is being recovered. In gase-phase reactions, palladium usually is supported on alumina to extend the catalyst life.

In jewelry, palladium hardened with 4-5% ruthenium provides a light, white, strong, tarnish-free alloy for watch cases, brooches and setting for gems.

Owing to palladium's ability to absorb and retain hydrogen, the latter diffuses selectivity through a palladium septum yielding pure gas. An alloy of palladium with 25% silver is the choice for this application because this alloy minimizes the disruptive dimensional effects that accompany a phase change in the palladium-hydrogen system.

A series of brazing alloys that contain palladium are utilized to join stainless steel, superalloys, and other high temperature alloys.

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*) Statistics on platinum-group metals uses and consumption in the USA for the period 1979 to 1983 (ref. 14)

A platinum-rhodium-palladium gauze catalyst and a recovery gauze reduce the loss of platinum in the production of nitric acid. A palladium-gold alloy is used for the latter application.

Rhodium Wrought rhodium has certain specific uses - in chemical processing (24%, including ammonia oxidation), in electrical (16%) and glass/fiberglass (16%) industries, in jewelry (6%) etc.- but the principal application of rhodium is as an alloying element for platinum. The 10%-rhodium-platinum alloy is used in the oxidation of ammonia, for spinnerets, glass-fiber bushings, furnace windings, and thermocouples. A trace of rhodium is required in the liquid bright golds used for decorating glass and porcelain to achieve a very fine-grained, bright deposit. Rhodium has been used in large quantities as the reduction catalyst for NO_x in catalytic converters. Small amounts of rhodium plus iridium or ruthenium are used in dental alloys to produce a very fine-grained casting.

At present the most important industrial application for rhodium is its use in automotive emission control devices, catalytic converters (38%).The increase in auto purchases in the USA during 1983 and 1984 was combined with the implementation of auto emission standards in Japan and the expectation that emission standards will be introduced in Europe by the end of the 1990's to produce both immediate shortages of rhodium and the prospects of even larger shortfalls in the supply of this metal in the future. An additional 1500 to 1900 kg could be required each year should Europe adopt new emission standards requiring the use of rhodium-bearing catalysts (as well as several tonnes per year of platinum and palladium).

Ruthenium The principal uses are in electrical (61%) and chemical (31%) industries, and jewelry (8%).

Titanium anodes coated with ruthenium oxides or platinum-iridium are employed in the chlorine and chlorate manufacturing industry in the electrolysis of brine. Such coatings are less expensive than graphite anodes.

Ruthenium is the principal constituent in resistive glazes in thick-film electronic circuits. Ruthenium-platinum and ruthenium-palladium are used in electrical contacts and in contacts for voltage regulators, thermostats, relays, gasoline gauges, and oil gauges. Ruthenium may be employed increasingly as an electroplating deposit in many light-duty electrical contacts.

Pure ruthenium can be worked to rod and strip at high temperature with great difficulty; it is often used as a hardener in the highly ductile platinum and palladium alloys.

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PART II
COUNTRY STATUS REPORTS

SPENT FUEL MANAGEMENT IN ARGENTINA

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Abstract

The current Argentine nuclear power programme consists of HWR reactors: two in operation (Atucha-I, 345 MWe and EMBALSE, 600 MWe) one 745 MWe is under construction and another one, 700 MWe will be installed before the end of the century. Plans for spent fuel storage and active programme for the utilization of Mixed Oxide (U-235, Pu-239) fuel which allows the development of technology for reprocessing and MOX fuel fabrication on a pilot plant are described.

1.GENERAL CONSIDERATIONS

Spent fuel management criteria at Argentina correspond to a nuclear program with HWR reactors : two in operation (ATUCHA-I, 345 MWe and EMBALSE, 600 MWe), one 745 MWe under construction and another 700 MWe to be installed before the end of the century.

So with only 4 or 5 Nuclear Power Plants (NPP) for the next 15 years, the following assumtions could be considered :

- In the next 15 years, no spent fuel storage or final disposal problems for the generated fission products could arise.

- No industrial capacity could be justified for back end of fuel cycle activities.

- Only research and development programs, pilot plants or demonstration programs could take place.

On the other hand, two NPP, the HWR pressure vessel type, have an expensive fuel element for the 6,000-7,000 MWD/T discharge burnup obtained. Also it was found that uranium reserves at Argentina are not large and with low uranium content (approx. 1 %.).

Based on those facts, alternative fuel cycles for these HWR reactors were considered which are briefly described in the next sections.

1.1 MIXED OXIDE FUEL UTILIZATION - PHASE I

It is well known that the discharge burnup of fuel elements in a heavy water reactor can be significantly increased by slightly increasing the content of fissile material (U-235, Pu-239) in the fuel. This results in a reduction of natural uranium consumption, the number of spent fuel to be stored in the NPP, the possibility to reuse spent fuel and in the case of using enriched uranium a considerable cost saving.

There are several ways by which these can be achieved . Investigations performed for Atucha I and Atucha II reactors show that a minimum of natural uranium consumption is obtained at an enrichment of about 1.2 % U-235. Similar results were found for CANDU type reactors.

The increase of discharge burnup for an equilibrium core with increasing enrichment for PHWR's is shown in Table 1, and also the direct annual fuel cost. A good alternative is the use of U-235 spikes during some time, and to replace them by MOX spikes when Pu is available from the reprocessing pilot plant. In this way seven spike fuel elements could be prepared, for a demonstration program.

Table 1 shows that there is an important reduction in uranium consumption in the case of Pu spiking compared to the natural uranium cycle. Fuel cycle economics could be obtained because only the spikes elements involve increased fabrication costs.

We have started the low enrichment of some ATUCHA-1 fuel elements with 0.85 % U-235. These fuels are already manufactured and a second lot will be prepared during 1986. A reprocessing pilot plant and a MOX fuel rod pilot plant are under construction.

TABLE 1

REACTOR CORE	ENRICHMENT w %	BURNUP MWD/t	ANNUAL U REQUIRED tU	ANNUAL COST mill.US$
ATUCHA I NPP :				
HOMOGENEOUS	NAT. U	6,000	58	19
	1.2	16,400	47	12
SPIKE U-235	2.0(60)	23,600	43	12
	NAT.(193)	10,400		
SPIKE Pu-239	2.0(60)	26,300	25	
	NAT.(193)	11,400		
ATUCHA II NPP				
HOMOGENEOUS	NAT. U	7,500	85	27
	1.2	20,600	69	21
SPIKE U-235	2.0(108)	24,600	72	20
	NAT.(343)	11,600		
SPIKE Pu-239	2.0(108)	28,000	42	
	NAT.(343)	12,200		

Note: Numbers between brakets indicate number of fuel elements in the core.

The program described on phase I will allow the development of technology for reprocessing and MOX fuel fabrication, as well as the experience on the use of enriched fuel in HWR at a pilot plant scale.

1.2 MIXED OXIDE FUEL UTILIZATION - Phase II

It could be foreseen that after the year 2010, industrial reprocessing could be needed for a breeder program or because of problems related to radioactive waste final disposal.

The technology developed during phase I will not be suitable for an economic and efficient operation of industrial plants. It is advisable to have a second phase in the program on a semi industrial scale. For instance, if we consider the two ATUCHA power stations operating with Pu spiked cores, a 100 tU/year reprocessing plant would be required, as well as the corresponding waste treatment plant, waste disposal site, and MOX rod fabrication line. A considerable percentage of the investment could be recovered from fuel sales, reducing technology development costs.

2. SPENT FUEL AND WET STORAGE HISTORY

2.1 SPENT FUEL HISTORY

2.1.1 ATUCHA I NPP

Fuel discharged from the reactor has been stored at the NPP since 1974. Fuel elements have 36 rods with Zry-4 cladded UO2 pellets active length 5.3 m. Total U per fuel element is 152.5 kg. The total number of fuel elements in the pools at Feb. 1986 is 4,076 with a calculated Pu content of 1,886 kg.

The discharge burnup is about 6,000 Mwd/t. In 10 cases the discharge burnup reached more than 10,000 Mwd/t.

One fuel element was severely damaged during operation and was encapsulated. Other 35 fuel elements had had different types of damages but didn't need to be encapsulated. One fuel element was damaged during its handling in the pool area and two during loading operation in the reactor.

About 100 discharged fuel elements were introduced in the reactor in order to increase burnup with good results.

Annual fuel consumption for a 0.8 load factor is 57.4 tU .

2.1.2 EMBALSE NPP

Spent fuel has been stored since 1984 .

Fuel elements are CANDU type, 37 rods, Zry 4 cladded. Total U per fuel element is 19 tU.

About 6,565 fuel elements were discharged since start up of the NPP in 1983 until 1/31/86, with an average burnup of 6,610 Mwd/t.

One encapsulated defective fuel element is in the pool. Other fuel elements discharged because of probability of leaks, are not encapsulated.

Total Pu in the pool is 415.6 kg.

Annual fuel consumption for a 0.8 load factor is 88 tU.

2.2 POOL HISTORY

2.2.1 ATUCHA-I NPP

Pool storage capacity was increased from the original 10 years pool house (two pools), to 15 years with a second pool house (four pools). All pools have stainless steel lining.

The water ph is 7, and its temperature is 35 C. No boron or lithium is added to the water. A diatomea filter was used but it showed problems during operations. In the second pool a resin filter was developed and used with good results.

No biological growth has taken place.

Equipment for post-irradiation examination of fuel elements was installed. No mayor problems have been found during the 12 years of operation of the pools.

2.2.2 EMBALSE NPP

The present pool storage capacity is 15 years. Pools have expoxi lining. In order to perform post irradiation examination of fuel elements, equipment has being installed, including disassembling of rods for cladding inspection.

3. IRRADIATED FUEL TRANSPORT

The container of irradiated fuel elements responds to the needs for transportation of the fuel elements from the Atucha I Nuclear Power Plant to the Ezeiza Research Center, 110 km apart.

The following data were considered for its design :

- Capacity : 1 fuel element
- Burnup : 6,000 Mwd/t
- Decay time : 6 years (minimum)
- Shielding : steel

The equipment consist basically of a low carbon steel cylinder of approximately 400 mm external diameter and 6,500 mm of length. A changable canister is located inside with an internal diameter of 110 mm. The cask is closed in both ends with low carbon steel lids. The canister will have its own internal lid as a first containment for the fuel. In all cases there will be joints and/or ring-seals to assure an effective closure of the lids.

4. REPROCESSING PLANT

A plant for the reprocessing of nuclear fuels was designed and constructed by the Argentine Atomic Energy Commission. The plant is located at the Atomic Center at Ezeiza, approximately 50 km south-east of Buenos Aires.

The plant will have an annual throughput of 5 metric tons of uranium, on the basis of 200 operating days per year.

The program objective is : a) to obtain experience to be used in the construction and operation of a larger reprocessing plant and b) the production of fissile material for the mixed oxide demonstration program.

Uranium oxide fuel elements irradiated to an average of 6,000 Mwd/t and cooled for several years, will be the first fuel to be reprocessed. The storage pond will provide buffer capacity and allow appropriate cooling.

4.1 TECHNICAL BASIS AND DESIGN PRINCIPLES

In the head-end part of the plant, the fuel elements will be chopped in a horizontal feeding machine and leached with nitric acid to dissolve the fuel.

Capabilities for gamma scanning of leached hulls will be provided at the head-end cell.

Solvent extraction has been chosen for the separation of the nuclear fuel and fission products. Plutonium and Uranium will be coextracted, separated and decontaminated in a three solvent-extraction cycle using 30% TBP/OK.

Air pulsed mixer-settlers have been chosen as contactors. Plutonium stripping with ferrous iron will be adopted. Final decontamination and purification of plutonium will be made using anion exchange resins.

The final products of the process will be a solution of uranyl nitrate and plutonium dioxide powder.

Uranium-Plutonium mixed oxides will be fabricated in a demonstration scale in appropriate rooms of the reprocessing plant.

The maintenance techniques chosen for the reprocessing plant was the direct course one. This choice was based on the small plant capacity and costs. However, some areas of plant process have been provided with remote and semi-remote maintenance techniques, such as the head-end cell and restricted operational areas.

For a better appreciation of the plant, Figure I shows and describes the most relevant areas.

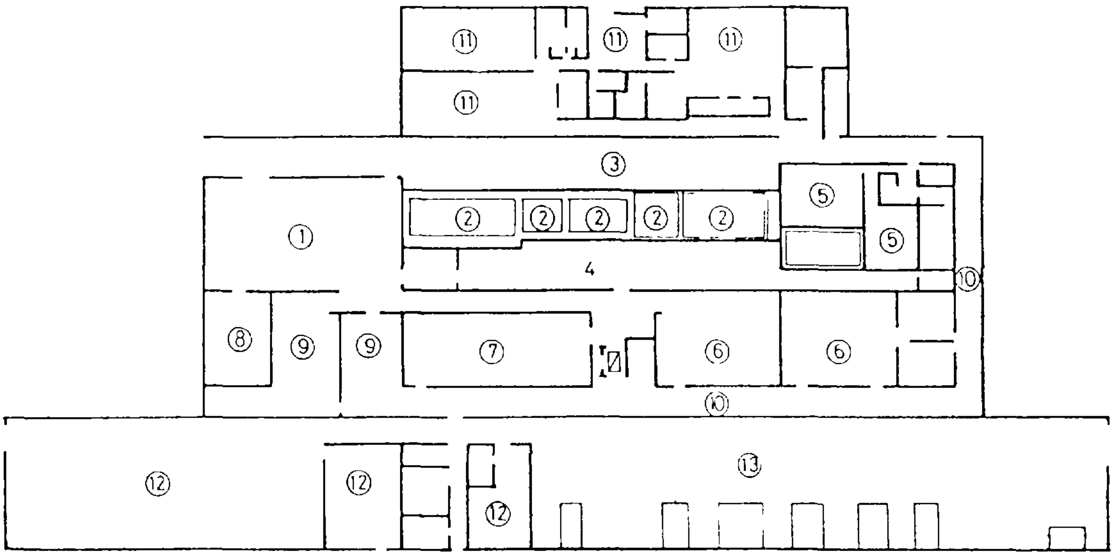

Fig.1. Reprocessing plant.

At present time the plant's construction is 80 % completed.

Cold tests are expected to be finished on 1988 and tests with non-irradiated U and Pu during 1989.

4.2 STORAGE AND TREATMENT OF RADIOACTIVE WASTES

4.2.1 LIQUID AND SOLID WASTES

The liquid radioactive waste produced in the plant, excluding the high activity fraction (hwl), will be subjected to a process of concentration and subsequent storage. For example the depleted uranium solution will be concentrated until they reach crystallization and then stored in drums.

The solid waste, depending on the level of activity and its nature, will be stored in concrete. It is estimated waste storage in the plant for some 6 years of operation. In this plant the final waste treatment will not take place.

5. MIXED OXIDE FUEL DEVELOPMENT

A mixed oxide fuel rod laboratory has been operating since 1975, for the development of technology and personnel training.

A new facility is now planned and will be located at the reprocessing pilot plant with a capacity according to the Pu availability from such plant. The facility will handle up to 5.5

m rods for Atucha I fuel elements. The facility will also include alpha active waste treatment , liquid waste precipitation and solidification, and low level solid waste by hot pressing and subsequent inclusion in concrete. The fuel rods assembly work will be performed at the fuel element manufacturing company located at the same site.

6. FINAL DISPOSAL

The Argentine nuclear programme intends for the reprocessing of spent fuel elements and recycling of plutonium produced for the generation of electricity as was described in previous sections. As a result, high-level radioactive waste initially arise in liquid form and will then have to be disposed of by appropriate treatment in the long term. The basic protection criteria applied for this purpose are: to avoid non-stochastic radiation hazards, to limit the risk to the individual and to reduce the collective radiological impact as far as is reasonably achievable. On the basis of these criteria, the selection of a site and the basic engineering project for the construction of a repository to meet the country's needs have been undertaken. In the first stage, some 200 potentially suitable granite occurrences distributed throughout the country were identified.

A preselection then was made of granite formations which lie outside seismic areas, which are parts of stable and scarcely altered geological formation, and which also lie outside potential mining areas. Four granite outcrops were identified in this preselection as being the most promising for further study. Detailed investigations were performed in one of the preselected granite formations which is located in the province of Chubut in southern Argentina.

SPENT FUEL MANAGEMENT IN BELGIUM

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Abstract

Belgium currently has 5.5-7 GWe installed capacity with another 1-1,4 GWe planned but not yet committed. The spent fuel is stored at reactor site in storage pools before being sent to a reprocessing plant.

The plan for spent fuel reprocessing and recycling of fissile products recovered into thermal and fast breeder reactors and contracts existing with COGEMA for the reprocessing of approximately 600 tonnes U of fuel unloaded from Belgian power plants up to 1989 are outlined.

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1. PROGRAMME STATISTICS

	1985	2000
Installed nuclear capacity (units -- GWe)		
-- connected to the grid	7 - 5.5 GWe	8 - 6.9 GWe
-- under construction	-	?
-- planned but not yet decided	1 - 1.4 GWe	?
Nuclear electricity share of total primary energy (percentage)	16.4	22.3
Electricity share of total primary energy (percentage)	29	38.5
Nuclear share of electricity (percentage)	56.5	58

2. PROGRAMME PLANS

2.1 Nuclear generation

2.1.1 Key factors which could affect projected capacity to 2000

The national electricity generation development programme, which is drawn up by Belgian electricity producers at regular intervals, must be approved by the Secretary of State for Energy prior to implementation.

In view of the time required for completing the administrative licensing procedures and also for constructing the generation facilities, the period covered by the latest development programme stretches from 1985 to 1995.

This plan is governed by several factors, the more important being:

-- Foreseeable trends in electricity demand;

-- Security of supply to customers;

-- Cost per installed kW depending on the different types of plant that might by envisaged and relative capacities;

-- Current and foreseeable fuel costs.

Electricity demand forecasts are a key component of the model. By means of scenarios for macroeconomic and energy trends, a coherent set of possible trends in electricity consumption is evaluated.

In the 1985-1995 development plan, electricity demand in terms of energy as well as capacity is defined as following three exponential trends of 1.5, 2.5 (most probable rate) and 3.5 per cent per year, respectively.

Several factors influence any consumption forecast, namely:

-- Prospects concerning the economic situation during the next few years;

-- Impact of replacing certain energy vectors by electricity, e.g. penetration of electrical heating in the domestic sector;

-- Improved energy yields of electrical appliances and processes;

-- Outcome of the rational use of energy policy in the industrial, domestic and tertiary sectors.

For a growth rate of 2.5 per cent per year in electricity demand, the generation facilities required for the period 1985-1995, in addition to the 25 per cent Belgian participation in the two French nuclear plants at Chooz of capacity 1 390 MWe each, would be as follows:

-- A 1 390 MWe nuclear plant to be brought into service in 1995, with possibly a 50 per cent French participation;

-- A 105 MWe fluidised bed unit to be brought into service in 1990 and using normal coal;

-- A 105 MWe fluidised bed unit to be brought into service in 1995 and burning waste coal or possibly gas turbines (or gas turbine-steam turbine generator sets), depending on coal price trends.

In view of the different lead times, no decision is to be taken at the moment on the last two units.

Apart from energy trading between Franco-Belgian nuclear plants, the development plan does not take into account any other electricity exports or imports. This obviously does not preclude the fact that co-ordination exchanges between the electricity grids of neighbouring countries and Belgium might arise in future as in the past.

2.1.2 Plan beyond 2000

The latest electricity generation development plan submitted by Belgian producers to the competent authority for approval does not extend any further than 1995.

At this stage it is not yet possible to give a reliable assessment of the nature and capacity of generation facilities to be brought into service beyond that date.

2.2 Spent fuel management

2.2.1 Interim storage

By 1986, approximately 300 spent fuel assemblies containing about 140 tonnes U will be unloaded each year from Belgian nuclear plants.

The spent fuel assemblies will be stored at the reactor site in storage pools for at least one year before being sent to a reprocessing plant. Total net capacity of storage pools at Belgian plants (Doel 1, 2, 3 and 4 and Tihange 1, 2 and 3) is about 2 460 assemblies, possibly extending to about 740. This capacity does not include a reserve allowing the possibility of unloading the entire core for each unit.

The spent fuel storage pool linked with the ex-Eurochemic reprocessing plant, which is now shut down, could be repaired and receive about 800 fuel elements from Belgian power plants.

2.2.2 Plans for ultimate disposal

a) Reprocessing

Like other industrialised countries with no uranium resources of their own, Belgium reprocesses spent fuel with a view to recycling fissile products recovered from thermal and fast breeder reactors.

Through SYNATOM, Belgium has negotiated with COGEMA contracts for reprocessing approximately 600 tonnes U of fuel unloaded from Belgian power plants up to 1989.

As for possible recycling of spent fuel in Belgium, it should be noted that in 1984 SYNATOM set up a specialised subsidiary, BELGOPROCESS. Initially, Belgoprocess will operate the ex-Eurochemic nuclear site at Dessel and carry out the tasks which Belgium has undertaken to perform on behalf of Eurochemic. A decision was expected by the end of 1985 regarding the possibility of restarting the ex-Eurochemic reprocessing plant.

This decision has been delayed and negotiations with foreign potential partners have been resumed in order to restart of the reprocessing plant in the frame of an international collaboration.

b) Disposal of high level waste

CEN/SCK is currently conducting a research and demonstration programme at the Mol site concerning the disposal of conditioned high level waste and plutonium waste in deep-lying clay beds.

The programme started in 1974 and provided an opportunity for a fully detailed study on the technical feasibility of the project and intrinsic safety factors.

Clay acts as a safety barrier in two ways: it is impermeable to water and with the other soil components, it forms a barrier against radioisotopes through a number of phenomena, such as ion exchange, precipitation and chemical reduction.

In 1984 the construction of an underground laboratory at a depth of 225 metres in the clay formation was completed. The laboratory is used for in situ tests on materials corrosion, leaching, heat dispersal and radioisotope scattering.

The second part of the programme, which is still at the preparatory stage, consists in building a larger underground chamber in order to test and study drilling and alignment technologies, mechanical problems and radiation effects etc.

The next stage will consist in constructing a demonstration gallery along with storage wells and handling equipment. The demonstration includes the handling and placing of conditioned waste.

Since 1983 through the Organisme national des déchets radioactifs et des matières fissiles (ONDRAF/NIRAS) potential users of the clay disposal technique have been partly financing the expenditure connected with the implementation of the above R & D programme, in particular the share of expenditure not met by the EEC Commission.

ONDRAF/NIRAS is also currently preparing a preliminary safety and feasibility report to be submitted in 1987 to the competent authorities to enable the latter to take a decision on the construction of a clay underground repository so that the disposal of conditioned high level and long-lived waste can be performed in due course under optimum safety conditions.

2.3 Low and intermediate waste management

2.3.1 Current situation

The main waste producers, including nuclear power plant operators, condition a large share of their own waste in situ. The remainder and the waste from other nuclear activities (fuel fabrication, research, production and use of radioisotopes etc.) are transported to central processing and conditioning facilities operated by the Nuclear Study Centre CEN/SCK at the Mol nuclear site.

After conditioning, all the waste is collected at storage facilities in Mol prior to ultimate storage.

As for the disposal of low-level conditioned waste, Belgium has participated at regular intervals in the dumping operations in the Atlantic organised under the Nuclear Energy Agency's multilateral mechanism. After the dumping operations were interrupted in 1983 on technical grounds, the conditioned waste was stored again at Mol.

The Organisme national des déchets radioactifs et des matières fissiles ONDRAF/NIRAS is responsible for organising the different stages in the system described above.

2.3.2 Future plans

While still in favour of dumping low radioactive waste under international surveillance, Belgium is currently developing an alternative consisting in the development of a land disposal site within the next ten years.

Under this alternative programme ONDRAF/NIRAS carries out the following activities:

-- Improvement of conditioning techniques with a view to cutting right down the volume of waste to be stored and disposed of; the high temperature incinerator developed and constructed by the Nuclear Study Centre CEN/SCK is an effective means of meeting this objective;

-- Construction of a storage building for storing conditioned waste until disposal becomes possible;

-- Study of a surface or shallow-lying land disposal site and development of this project after a licence has been granted by the safety authorities.

3. AREAS OF WORK WHERE INTERNATIONAL CO-OPERATION AND/OR JOINT STUDIES ARE OF INTEREST

The following fields are of special interest to Belgium:

3.1 Questions relating to advanced thermal reactors: increased conversion factor, increased burn up, increased safety.

3.2 Questions relating to fast breeders. International co-operation in this field is already operating.

3.3 Plutonium recycling. Studies on materials and core management.

3.4 Conditioning and disposal of radioactive waste. Possibility of including other partners in the programme run by CEN/SCK on waste disposal in clay formations.

3.5 Simplification of procedures connected with the nuclear fuel market (purchasing, transport, safety).

SPENT FUEL STORAGE IN CZECHOSLOVAKIA

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Abstract

Czechoslovakia currently has six WWER-440 reactors operating and six more units at different stages of construction. It is planned that all 12 units will be operational by mid-1990. In accordance with the general bilaterial agreement with the USSR on the cooperation in nuclear development, the Soviet Union will accept the return of spent fuel for reprocessing. An outline of cooling pools for spent fuel storage at different time periods is given.

Electricity generation based on the nuclear source was started in Czechoslovakia on an industrial scale in 1979 when the first PWR-unit of the WWER-440 type was put into operation at the nuclear power plant Jaslovské Bohunice. In pursuance of the established near-term programme of nuclear power development, comprising a total of twelve such reactor units to be built at three power plant sites, the present status is represented by six reactors in operation /4 at Jaslovské Bohunice and 2 at Dukovany/ and six remaining units /2 at Dukovany and 4 at Mochovce/ at different stages of their construction. It is expected that the number of operating reactors will increase to ten in 1990 and to twelve a couple of years afterwards. In the next period, the WWER-1000 units will be introduced.

Concept of arrangements related to the nuclear fuel cycle and its back-end in particular has been motivated by conviction prevailing in early 1970s as for the most likely trends in nuclear power progress during the next three decades. It was decided to adopt the option of spent fuel reprocessing with a following recycling of the recovered uranium and plutonium in fast breeder reactors whose commercial maturity in the mid 1990s was considered for granted.

Within the framework of a general bilateral agreement with the USSR on the cooperation in nuclear power development, a clause fixing the compliance of Soviet Union to accept for reprocessing all spent fuel arisings from the WWER nuclear power plants was negotiated. The capacity of at-reactor cooling pools was designed for storage of three years' operating discharges and because the USSR was in possession of appropriate means for the spent fuel transportation including the casks, the ensured main technical provisions and services were held as adequate for spent fuel management in the near-term future. Also all requirements pertaining to the legal aspects have been settled in a satisfactory manner and the activities in this field have been put under the control of IAEA safeguards system.

The original assumption on the introduction of fast breeder reactors got modified in the beginning of 1980s when delays relating to the former timing became evident. This fact brought about a postponement of large-scale reprocessing needs with a consequential call for a revision of initial conceptions concerning the spent fuel management. Conclusions resulting from technical and economic evaluations led to the approval of a scenario retaining the reprocessing option and seeking for an adaptation of storage feasibilities in order to meet the new circumstances.

The least expensive, time consuming, and technically pretentious solution of the latter problem has been found in installation of away-from-reactor storage facilities colocated with the nuclear power plants. Their capacity has been assessed taking into account the annual spent fuel discharge of 14 MTHM from a reactor unit and conceiving a storage period of 10 years long enough to cope with all expectations of storage needs including an appropriate reserve. The proven technology of wet storage in borated water pools and the available USSR design of a storage facility for 600 MTHM in spent fuel have been chosen for this purpose.

Actually, such an away-from-reactor storage accommodation is in a very advanced state of its completion at the Jaslovské Bohunice site and is going to be put into operation in autumn this year. Designs of similar facilities at the both other nuclear power plant sites are elaborated.

The recent negotiations with the USSR and within the Council of Mutual Economic Assistance indicate that expedition of spent fuel for reprocessing after five years of its cooling is not excluded. A simple comparison between spent fuel arisings and the storage capacities disposable at the end of 1986 shows that the intermediate storage requirements of all three nuclear power plants can be covered by the existing away-from-reactor facility at Jaslovské Bohunice if the five years' storage comes into practice. Even if the initial concept of ten years is retained the intermediate storage capacity remains fairly satisfactory till the first half of 1990s. Therefore licensing of the away-from-reactor facilities for the sites of Dukovany and Mochovce is deferred and a revision of spent fuel storage arrangements is pending till 1990. As an alternative to the construction of new facilities in the future, the technique of compact storage is studied.

It is possible to state that no technical difficulties in spent fuel management and storage pool operation have appeared as yet. Perhaps a deal of this positive experience might be attributed also to the fact that we did not have to handle a single defective spent fuel assembly. Chemistry and radiochemistry values of the pool water have been kept according to the established parameters. No degradation of the spent fuel has been observed.

SPENT FUEL MANAGEMENT IN FRANCE

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Abstract

France's programme is best characterized as a closed fuel cycle including reprocessing and use of breeder reactors. It has 38,958 MWe installed capacity with another 21.010 MWe capacity under construction. The chosen spent fuel management scheme is that of reprocessing.

National programme and experience on storage, reprocessing and transport of spent nuclear fuel and contracts for reprocessing of spent fuel from different countries are described.

Spent fuel management in France is characterized by the reprocessing option; reprocessing being one of the major nuclear fuel industries developed to keep pace with the upgrowth of the national nuclear power program based on light water reactors and subsequently on fast breeder reactors.

After creating a first-rate uranium mining industry, an 11 million SWU enrichment facility at Tricastin and the necessary fuel manufacturing industry, France has launched two industrial projects which will raise the annual reprocessing capacity of La Hague facility to 1600t of water reactor irradiated fuel, based on reprocessing experience and R&D work :

- the UP3 plant (800t/year) which will be commissioned in 1989
- the UP2 800 plant, an extension of the existing plant, which will be commissionned in 1991.

R&D work is also in progress, together with the extension of the Marcoule Pilot Plant (TOR project), and preparations made for the industrial reprocessing of fuel from the Super Phenix reactor and the first fast breeder reactors (MAR 600 project).

An appropriate structure has been set up inside CEA Group to carry out the French industrial reprocessing program :

- COGEMA is the owner and operator of the plants;
- SGN (Société Générale pour les Techniques Nouvelles) is the prime contractor;
- CEA carries out the R & D work and acts as a "licensor" by providing both the above with the data necessary to design the process and the related equipment;
- ANDRA is responsible for waste management.

During the last years many communications have been made in international conferences and many publications has been issued : Some of them are indicated in the attached list of references and will be quoted and refered to in this working paper.

The importance of reprocessing from the energy conservation standpoint, the advantage of reprocessing from the waste management and safety standpoints and the economic aspects are presented in papers [2], [4] and [8].

1 - SPENT FUEL STORAGE AND TRANSPORT EXPERIENCE

The experience in the field of spent fuel storage and transport is globaly related to the number of nuclear power stations and the spent fuel reprocessing achievement; that's to say, the french experience is important in relation with the french nuclear program (65% of the electricity produced in France in 1985 came from nuclear power stations) and the La Hague reprocessing plant (nearly 2/3 of the spent fuel reprocessed in the world).

The French PWR program is illustrated by figure 1, with 44 units in operation and 17 units in construction. A spent fuel storage pool is associated to each PWR unit.

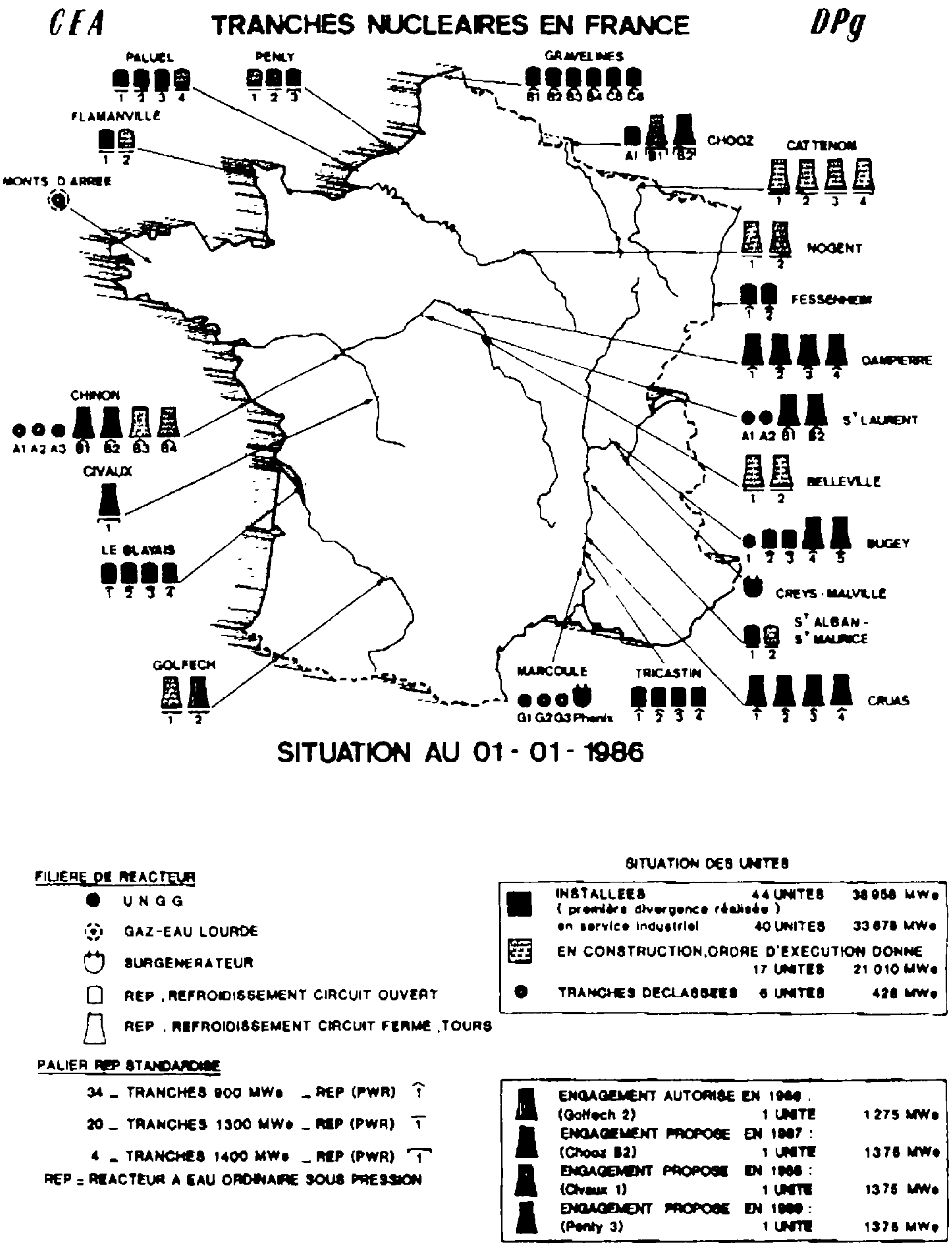
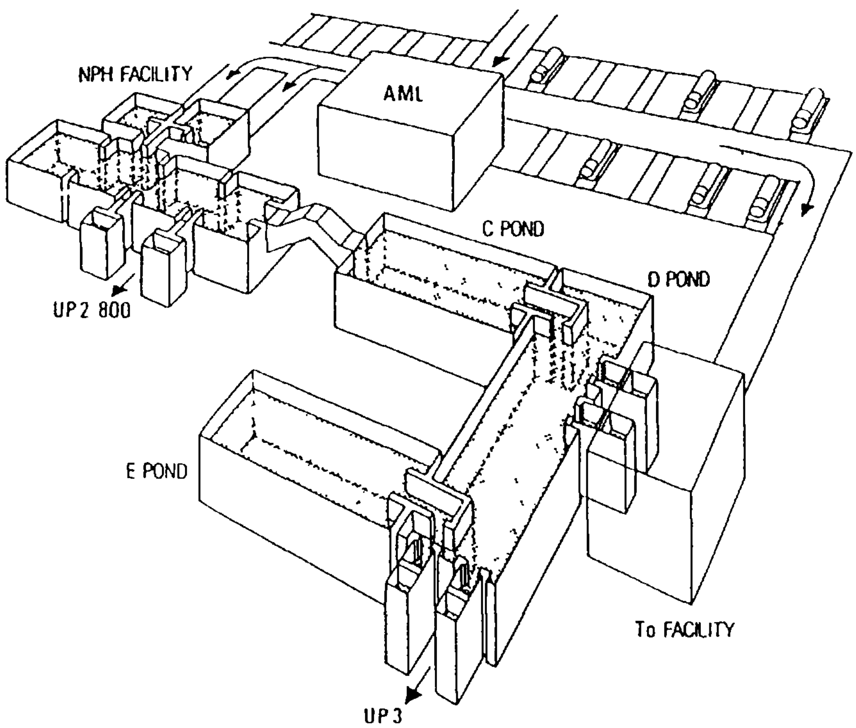


FIGURE 1



Reception and storage facilities planned at La Hague

FIGURE 2

The storage facilities of La Hague plant (figure 2) are described in paper [5] :
The first pond, with a capacity of 400t of uranium, through which transit the fuels intended to supply UP2/HAO was commissioned in 1975.

In view of the expansion programs currently under way, the need emerged to expand this storage capacity substantially by raising it to 8000t in 1988. Seven units have been or are being built :

. a receiving unit for road convoys which convey the fuels in their transfer casks to the La Hague facility. This unit, called AML, was commissioned in 1981. It is designed for buffer storage of the casks before their unloading or their shipment after unloading.
. An underwater unloading unit for the transfer casks (NPH), commissioned in 1981,nominal reception capacity 800 tU/Year.
. A fuel storage pond. This initial pond with a capacity of 2000 tU, forms part of the NPH facility. It was commissioned in 1981.
. A second fuel storage pond, Pond C, commissioned in 1984 (Capacity 2000 tU).
. A dry unloading installation for transfer casks, called "To" (800 tU/year) to be commissionned in 1986.
. Two additional storage ponds, Ponds D and E, similar to Pond C, operational in 1986 and 1988 (capacity 2000 tU each).

To achieve maximum flexibility in the operation of the installations, the storage ponds, unloading units and reprocessing plants will all be interconnected through the ponds.

The specific technologies developed for La Hague storage ponds are described in paper [4] :

- Nymphea bulb pond water cooling and treatment units
- PWR and BWR spent fuel storage baskets
- handling systems and shock absorber systems.

The experience accumulated through La Hague plant in the field of spent fuel transportation is presented in paper [6] and [22d]. It can be illustrated by a few figures updated end 84 :

- . transports of fuel from 50 power plants in France,Belgium, Germany, Japan, Netherlands, Sweden and Switzerland,
- . delivery at La Hague of more than 3400 tons of oxide fuel in about 1300 consignments.

Each year COGEMA manages directly or indirectly the transport to La Hague of approximately 300 casks either by road or by rail or by sea.
The flow of oxide fuel will increase by a factor of 2 in about 10 years.

Quantities of oxide fuel transported to La Hague

Tonnes U

1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
10	26	97	60	101	92	112	135	405	647	810	810

Two transports casks designers have succeded in obtaining COGEMA's approval and certification of their designs : TRANSNUCLEAIRE and LEMER. The basic data of casks described in [22e] and [22f] are summarized below :

	Loaded weight te	Capacity (fuel assemblies)	
		BWR	P W R
TN 17/2	72	17	6
TN 12/2	102	32	12
LK 100	102	-	12
TN 13/2	105	-	11 or 12

The main characteristics of these packagings are high safety and quality standards, large payload, moderate costs, reliability due to large experience, standardization facilitating fabrication, operation and spare parts supply, versatility (casks accepting different fuel of different types and characteristics including damaged assemblies).

As this time more than 70 of these casks are in operation and more than 20 under construction, most of them being of the Transnucleaire types.

In addition the TN 24 cask designed for intermediate storage of spent fuel is described in paper[22g].

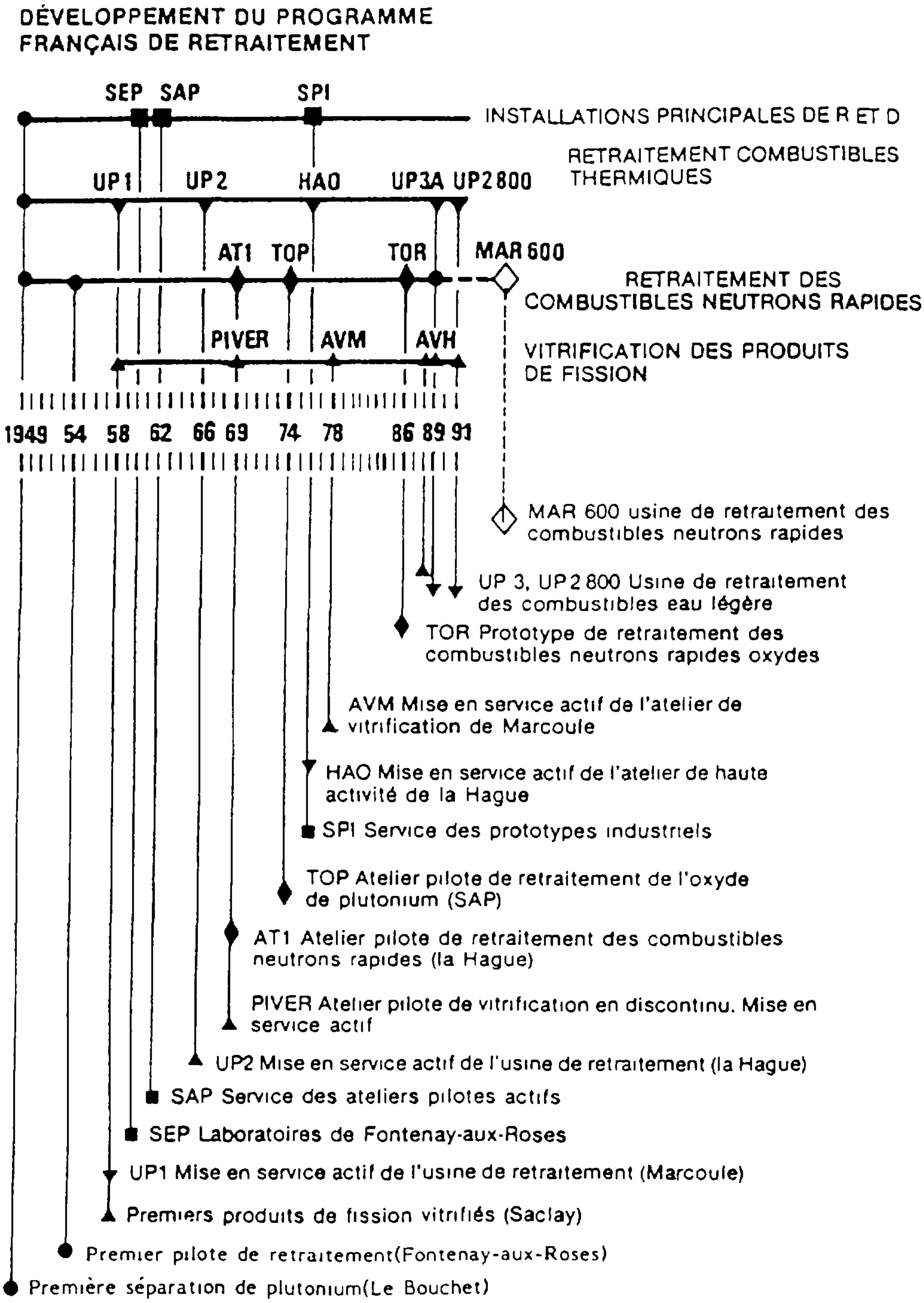
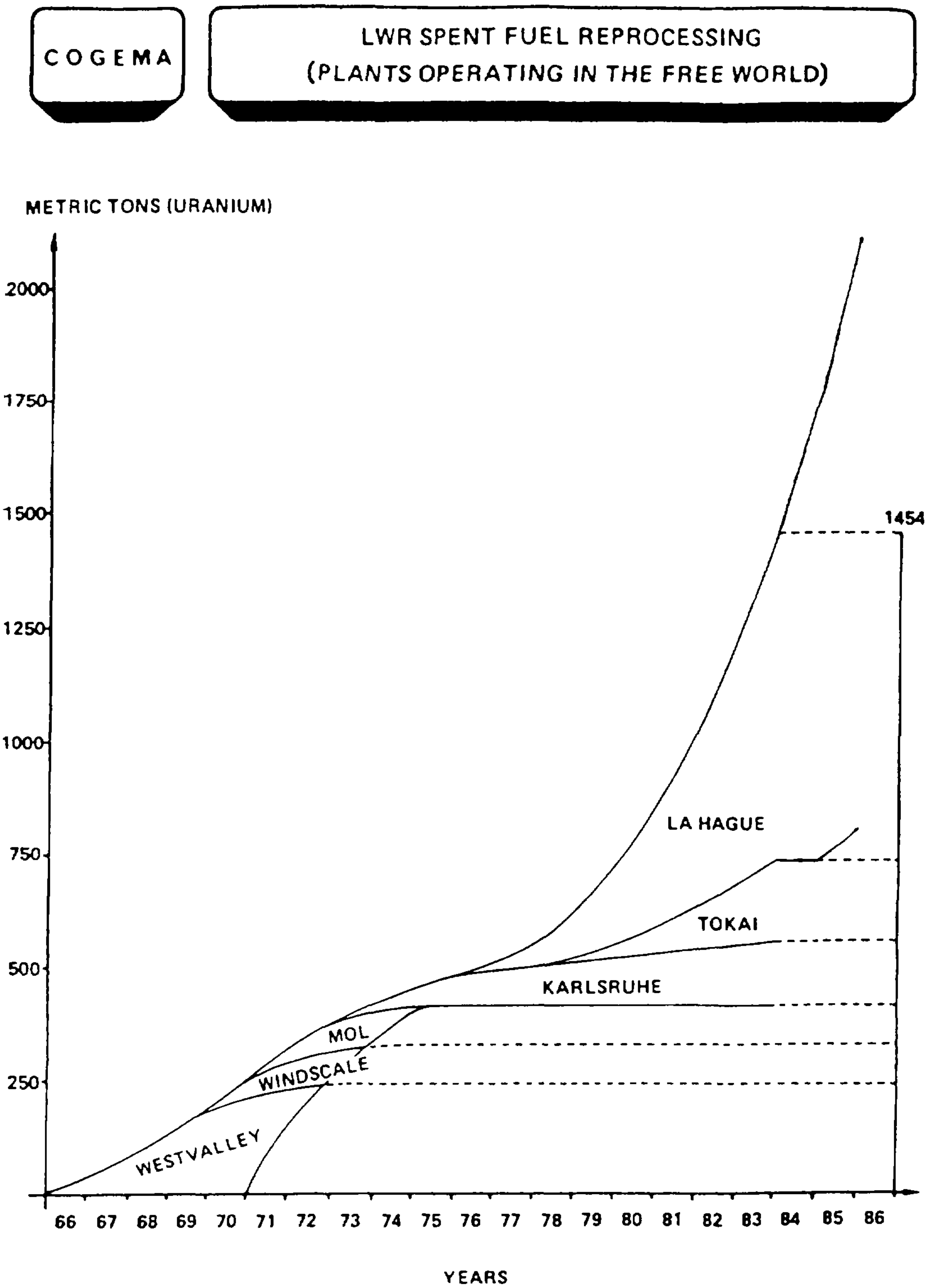


FIGURE 3

2 - THERMAL REACTOR FUEL REPROCESSING EXPERIENCE (figure 3)

UP1, the first French reprocessing facility was commissioned in Marcoule in 1958. It is still in operation after 25 years, with its expanded facilities, to reprocess graphite/gas fuels.

UP2, the second major reprocessing facility, built at La Hague, and initially intended to handle graphite/gas fuels, was commissioned in 1966. Up to now, UP1 and UP2 have reprocessed some 7000 tons of irradiated fuels from EDF and Spanish Vandellos gas-graphite reactors. Following the implementation of the pressurized water reactor system, it was decided to add to the UP2 facility a first oxide unit (HAO) to reprocess the irradiated fuels from light water reactors. HAO facility went on stream in 1976. By the end of 1985, HAO-UP2 plant had reprocessed 1338 tons of light water reactor oxide fuels (that is nearly 2/3 of the total tonnage reprocessed in the world as illustrated in fig. 4).

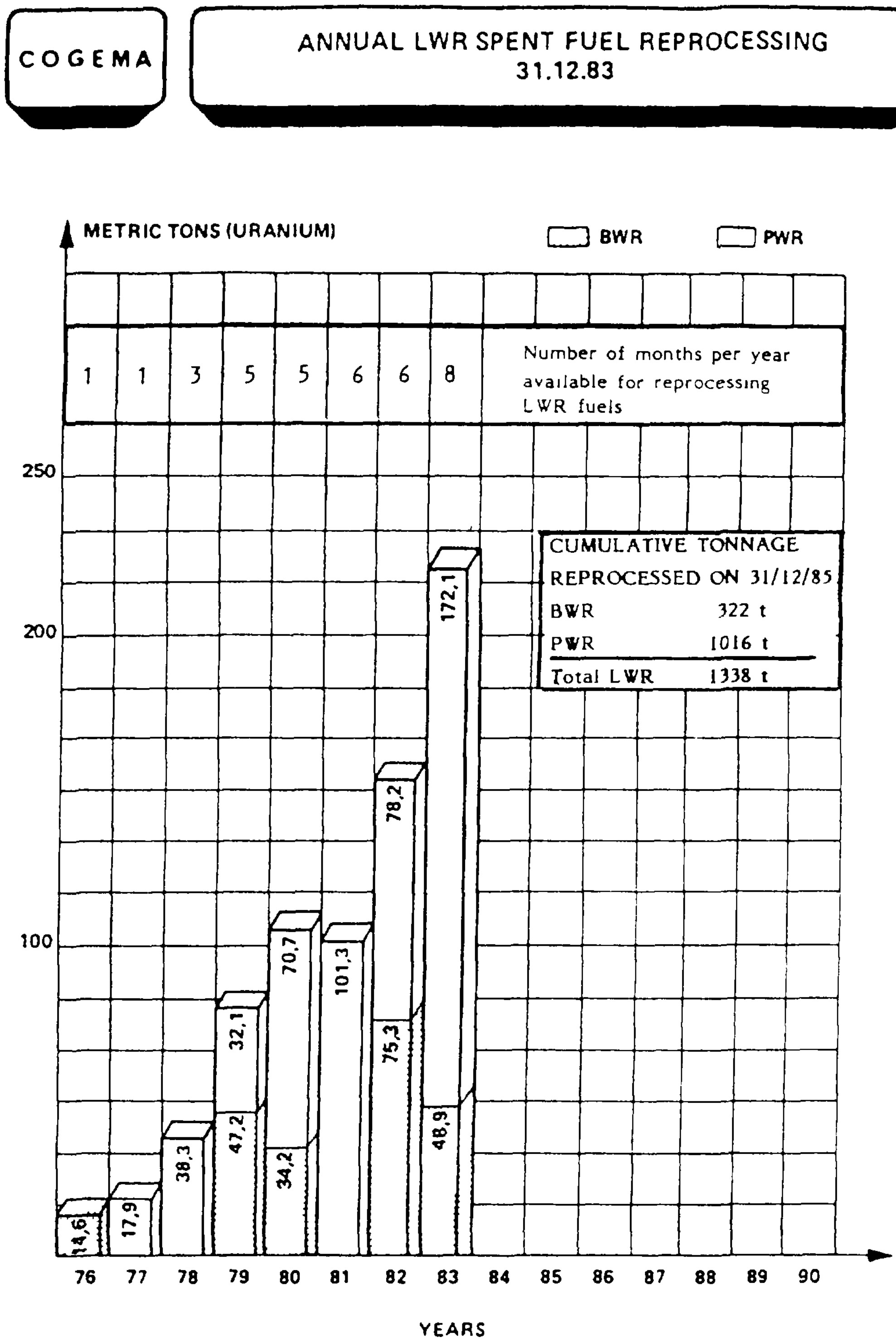


From C. AICOBERRY, COGEMA, JAIF March 1984 - updated end 1985

FIGURE 4

The HAO Project is described in reference [7] : basic data, operation procedure and essential equipment used, nominal capacity and performance.

Figure 5 gives the annual BWR and PWR spent fuel tonnages reprocessed from 1976 to 1983 and the number of months per year available for LWR reprocessing.



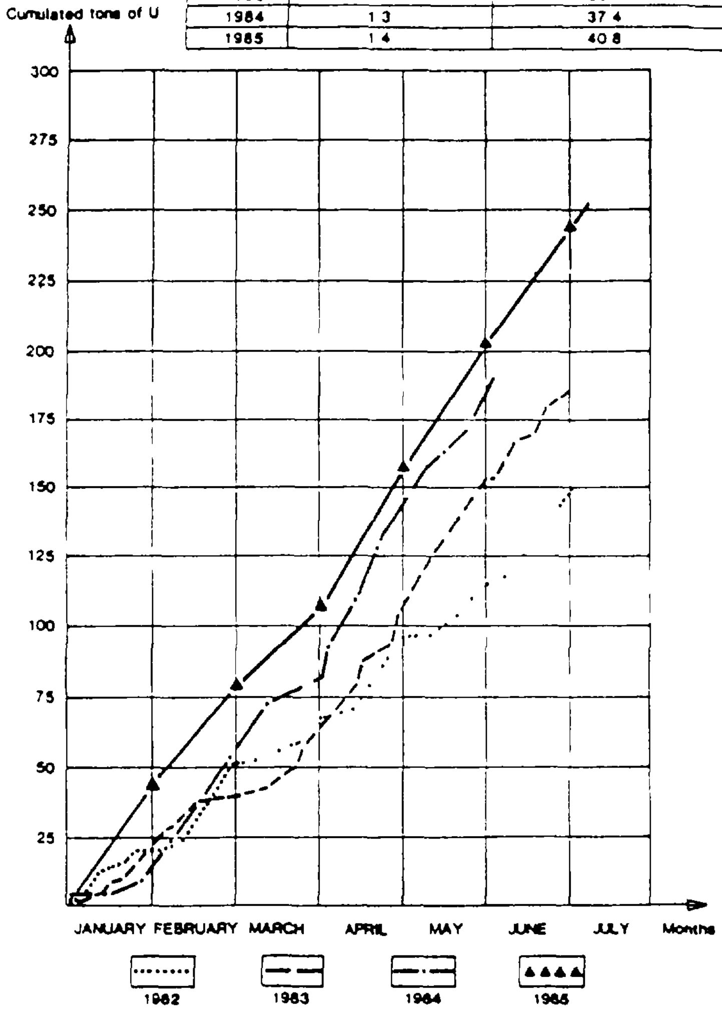
From C. AICOBERRY, COGEMA, JAIF March 1984

FIGURE 5

UP2 Plant operating performances are discussed in papers [7], [8], [9] and [10]. HAO maximum nominal work rate at the start of operations was set at four standard 500 kg PWR fuel elements per day, i.e., a maximum daily capacity of 2 MT, corresponding to a 400 MT/year capacity (300 reprocessing days per year with a load factor of 0,66). This capacity has been reached in passed years :
During the reprocessing run which began in November 1984 and ended in early July 1985, 316 tons were reprocessed, representing a monthly capacity of over 40 t, and corresponding to 400 t/year, for ten months of production. Figure 6, which gives the amount reprocessed in the first six months of recent years, shows this steady improvement in production. In terms of safety, the mean individual dose received by the personnel dropped from 500 mrem in 1975 to less than 200 mrem in 1984. This should be compared with the legal limit of 5000 mrem (see figure 7).

LIGHT WATER REACTORS REPROCESSING
(HAO SOUTH)

Year	Reprocessed tonnage / Working days	Average reprocessed tonnage / Month Up to 31st May
1982	1 0	23 6
1983	1 1	30 7
1984	1 3	37 4
1985	1 4	40 8

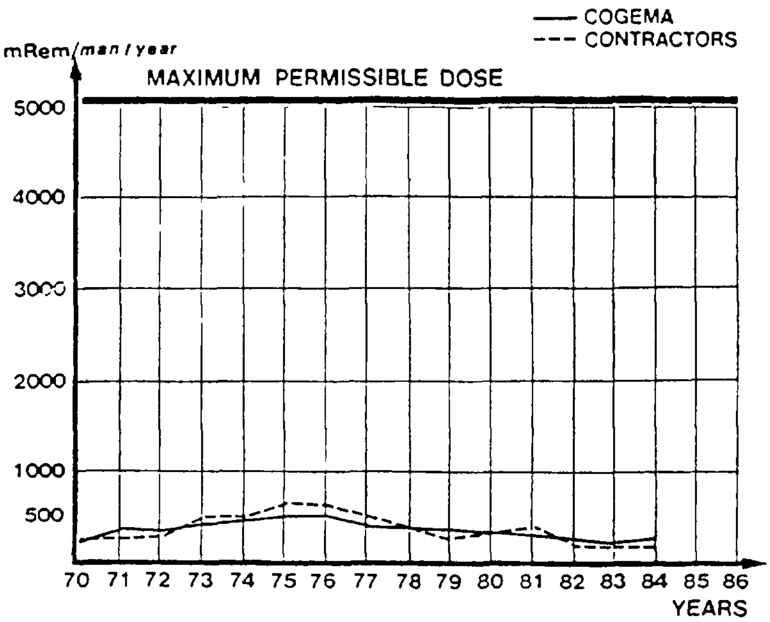


From Paper [8] J. COUTURE - G. LE BASTARD COGEMA August 1985

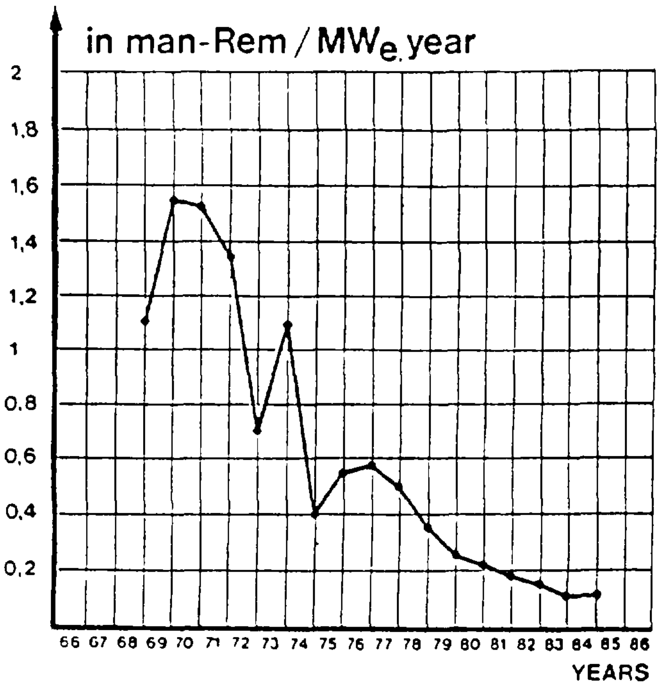
CUMULATIVE TONNAGE REPROCESSED DURING
THE FIRST SIX MONTHS OF EACH YEAR

FIGURE 6

AVERAGE RADIATION DOSES (12 MONTHS/FULL BODY)



SPECIFIC DOSES - COGEMA AND CONTRACTORS



From paper [8] J. COUTURE - G. LE BASTARD COGEMA - August 1985

FIGURE 7

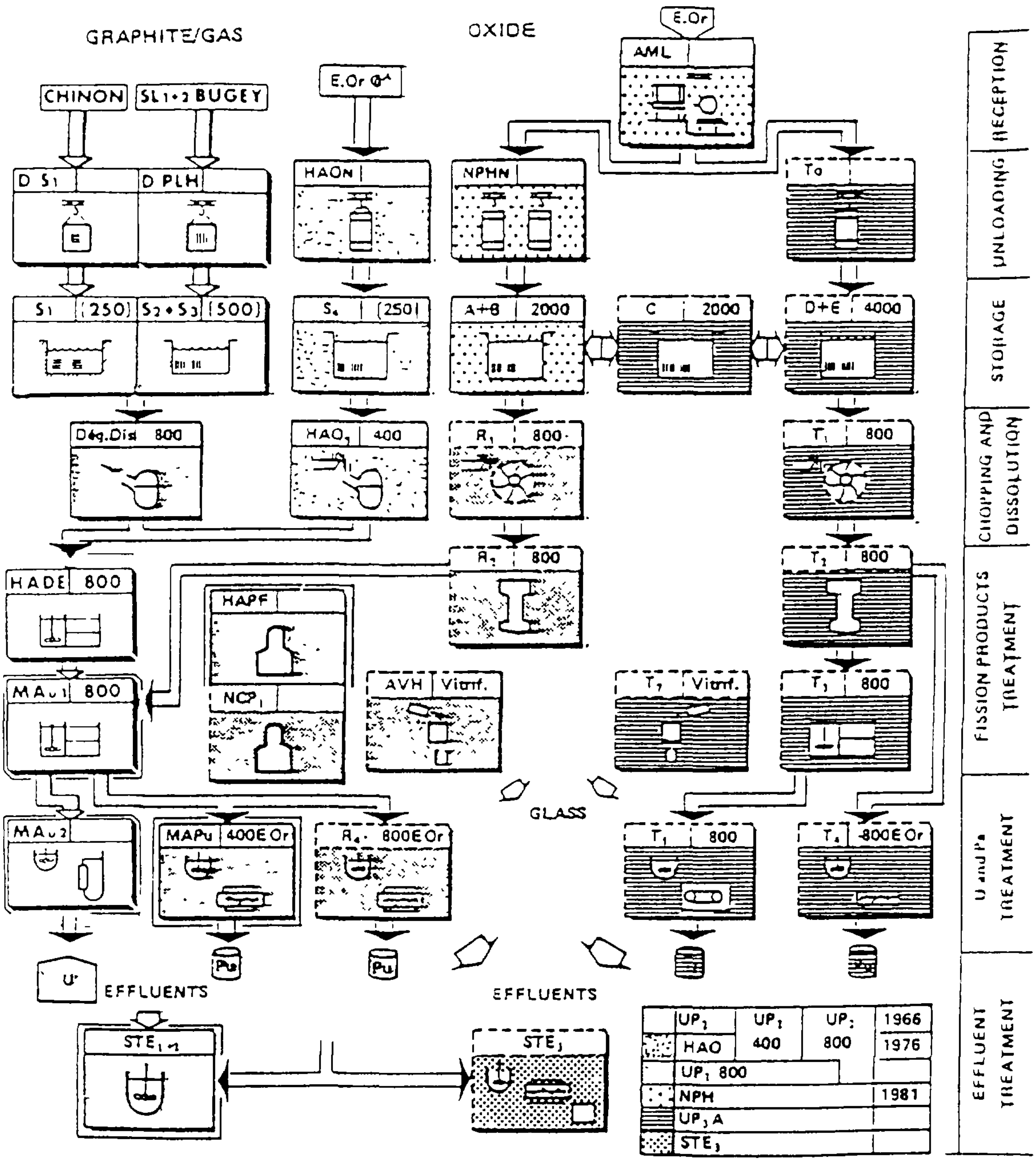
The specific dose (the ratio of the collective dose to the quantity of electricity generated by the fuel reprocessed) felt to 0.12 man-rem/MWe per year (see figure 7). This value should be compared with the 0.25 man-rem/MWe per year for uranium mines and nuclear power plants.

As for radioactive effluent releases, they are still far below authorized limits, despite the increase in tonnage reprocessed. Related to the quantity of electricity generated, activity released is steadily decreasing.

Paper [10] deals with last 25 years safety and radioprotection experience in reprocessing plants.

3 - EXPANSION OF LA HAGUE PLANT

Based on the first results obtained in the HAO-UP2 complex, which demonstrated the feasibility of industrial LWR oxide reprocessing, COGEMA decided to expand the La Hague facility significantly to achieve a capacity of 1600 tons/year in the 1990s. This expansion, which is currently underway, with the relevant installations shown in Figure 8 and described in paper [10 bis], will actually involve two plants, one representing an expansion of the present plant (UP2 800), and the second a completely new plant called UP3.



LA HAGUE PLANT
– Evolution of installations and capacities

FIGURE 8

These achievements will ensure the reprocessing of fuels from french and foreign utilities (EDF and thirty utilities belonging to six countries Japan, West Germany, Sweden, Switzerland, Belgium and the Netherlands) under contracts covering a total of 12,663 tons of fuels (including 6000 tons of foreign fuels intended for UP3). These contracts call for substantial financial participation by the foreign utilities in investments, and the return of the wastes to each of the countries.(Paper [10 bis] describes the reprocessing contracts).

The UP3 plant has a capacity of 800 tons/year and will be commissioned in 1989. Civil works were inaugurated in mid-1982. The UP2 800 plant will be based on the addition, to the present UP2 plant in operation at La Hague, of new units designed to raise its effective capacity to 800 tons of fuels per year also. The first new installations of this plant (new fission product concentration and vitrification unit in particular) are currently being built, but the construction of a new plant head end featuring a chopping/dissolution unit and a high-level extraction unit will only be undertaken subsequently. It is designed to reprocess also MOX fuels. The beginning of active tests of the complex is planned after that of UP3.
The head ends of both plants (UP3 and UP2 800) will be interconnected by a set of ponds designed to store 8000 tons of spent fuels as illustrated by figure 2. Associated with this complex is an effluent treatment station, waste decontamination and packaging facilities, interim storage facilities for packaged wastes, etc.

4 - FAST REACTOR FUEL REPROCESSING EXPERIENCE AND FUTURE PROSPECT

The french program for the reprocessing of fast neutron reactor fuels (fig.3) falls within the logical pattern of development of this system, in which the reactor has reached the industrial stage, passing through the three phases corresponding to the construction of the Rapsodie reactor, Phenix (250MWe) and Super Phenix at Creys Malville (1299MWe) which was coupled to the grid on January 14th 1986.

As soon as Rapsodie succeeded in supplying pins irradiated to significant burnups, laboratory tests were conducted at Fontenay-aux-Roses. These tests are still proceeding in so far as Phenix supplies fuels with increaseing burnups and different grades of clad materials.
The AT1 facility at La Hague, specially designed to reprocess Rapsodie fuel with a capacity of 1 kg per day, went on stream in 1969, and, by July 1979, when it was finally shut down, processed more than one ton of heavy metals derived from mixed oxides irradiated to 120,000 MWd t^{-1} (and sometimes only slightly cooled), thus guaranteeing the closure of the Rapsodie cycle several times.
The Marcoule pilot plant (SAP) adapted to reprocess Phenix and Rapsodie fuels since 1975, has handled 10.7 tons of fuels from Rapsodie, Phenix and the West German reactor KNK1. The plutonium recovered by reprocessing Phenix fuel served to close the cycle of the power plant, which presently contains about 140 assemblies fabricated with recycled plutonium. More than 75% of the fuel assemblies of the Phenix core are thus fabricated with recycled plutonium. Some of the fuels reprocessed actually contain plutonium that has undergone two passages in the Phenix reactor.

During these reprocessing runs of Phenix fuels at the SAP, described in paper [2], [12] and [13] excellent performance was achieved, both from the standpoint of plant capacity and that of safety and personnel irradiation. Adding this to the 9.9 tons of Phenix fuels reprocessing in the UP2 plant at La Hague, diluted with graphite/gas fuel, 21,7 tons (U + Pu) of fuels irradiated in fast reactors, including 16.8 tons (U + Pu) of Phenix fuels, have been reprocessed in France as illustrated in figure 9.

This unique experience has confirmed the feasibility of reprocessing fast neutron fuels, even at the highest burnups. It did not entail any significant change in the Purex process used for light water fuels, but allowed a set of improvements to this process and of waste characterizations, constituting indispensable know-how for passage to the industrial phase.

FBR FUELS REPROCESSING AT MARCOULE AND LA HAGUE (1965-1984)

REACTOR	Pu/U+Pu(%) core initial composition	MW day.t^{-1} max core	Cooling time (months)	QUANTITIES (t)
RAPSODIE FORTISSIMO	25 to 30	10 000 to 120 000	6-30	0,96
PHENIX (Pu)	18 to 25	30 000 to 100 000	10-40	16,83
PHENIX (U)	enriched U	38 000 to 45 000	10-30	2,3
KNK I (U)	enriched U	-	-	1,65
TOTAL				21,74

PLUTONIUM AVAILABILITY - SITUATION IN FRANCE -

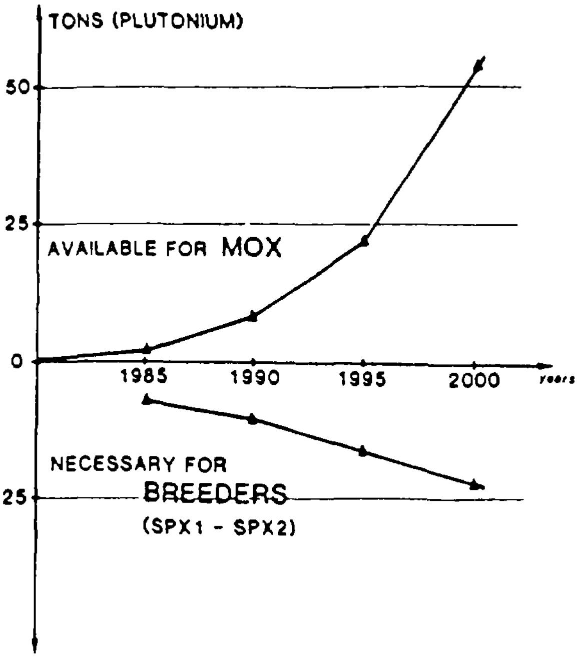


FIGURE 9

After having demonstrated the effectiveness of the process in significant conditions, an experimental program is currently proceeding with the aim of building industrial installations in the forthcoming decades. This program utilizes vast research capabilities, mainly concerning the reprocessing technology and systems for waste treatment, particularly the new TOR facility. This TOR (Traitement d'Oxydes Rapides) unit, launched in 1980, will be commissioned in 1986. It is described in paper [14]. It represents an expansion and renovation of the present Marcoule pilot plant (SAP) unit, carried out with two main objectives :

- to raise the capacity of the unit to 5 tons (U + Pu) per year, so as to guarantee the closure of the Phenix cycle, while preserving some availability for the experimental reprocessing of fuels from other fast reactors (KNK2, Super Phenix),
- to serve as a test bench for the new equipment designed for future plants.

Design work is currently proceeding on a preliminaty project for a small-scale industrial plant (50 t/year) called MAR 600 and corresponding to the reprocessing requirements of Super Phenix and two power stations of 1500 MWe each [paper 12]. This preliminary project is designed as an intermediate unit in the foreseeable medium-term development of the fast breeder reactor, with an attempt to secure the most favorable cost conditions. It features a single line of new equipment currently developed on full scale to reprocess 50 t/year of fuels characterized by a maximum burnup of 125,000 MWdt and minimum cooling time of three years. MAR 600 would be located on the Marcoule Center, where it will benefit from the existing infrastructures and general services (Laboratories, effluent treatment station etc..).

United Kingdom is also studying a 50t/year reprocessing plant project. Future choices for FBR reactors and FBR fuel cycle installations are to be made in the frame of european FBR cooperation.

5 - PLUTONIUM AND URANIUM RECYCLING

In addition to multiple recycling of plutonium in Rapsodie and Phenix fast breaders, Pu recycling is contemplated in PWR.
Figure 9 shows the use of plutonium resulting from EDF fuels reprocessing : Priority will be given to the satisfaction of FBR requirements (SPX 1 refuellings, supplied 51% by french Pu and the fuel of a second FBR, SPX 2 supplied in the same cônditions and presumed to be commissioned in 1995). The top part of the figure shows the remaining quantities of Pu that will be allocated to recycling in PWR's following the EDF decision of 1985 to recycle Pu in PWR for both economical and technical considerations.

MOX fuels fabricated by Belgonucleaire have been loaded in 1963 in BR3 reactor (Belgium), in 1968 in Garigliano BWR (Italy) and in 1974 in Sena PWR (France) and since then much cumulative experience has been built up for the handling of Pu and the fabrication of MOX fuels in France and Belgium.
Presently MOX production is carried out in two facilities : one in Dessel (Belgium) owned and operated by Belgonucleaire, the other at Cadarache (France) owned and operated by CEA where a LWR mixed oxyde line is added to those devoted to FBR fuels. COMMOX is the marketing joint-venture by Cogema and Belgonucleaire in charge of commercialisation of the total production of these two facilities. (Paper [8]).

The senario adopted for Pu recycling in EDF PWR calls for a first reload of MOX fuel bearing 8 metric tons in 16 fuel assemblies in a 900 MW PWR in late 1987.Fragema and Commox have set up arrangements to initiate this programme. The refuelling rate will rise to 90 metric tons in 1995, this corresponding to 10 x 900 MW PWR reactors recycling Pu. The expansion of fabrication capacities to meet the demand of the electrical utilities will be adapted optimally. This is why a large plant project is being investigated actively today. This project, of 100 to 150 t/year, called MELOX, could be located on the Cogema site at Marcoule, and could thus offer its industrial capacity from 1993.

Uranium recycling is also considered as illustrated in paper [24].
In 1987, EDF is to load experimental fuel assemblies containing uranium from reprocessing, fabricated by Fragema-FBFC.

Paper [25] gives an overview of plutonium and uranium recycling in PWR.

6 - RADIOACTIVE WASTE MANAGEMENT

The proponents of the nuclear option in France, recognized early that the success of the power program depends to a large extent on the level of safety and of reliability it achieves. This includes the safe long-term management of radioactive waste generated by nuclear reactors and fuel cycle facilities.
The french comprehensive waste management policy presented in paper [26a] is based on the acquired experience over 40 years through an extensive nuclear power program. French regulations are presented in paper [26b].
Adequate radioactive waste treatments are incorporated in irradiated fuel reprocessing facilities such as vitrification of high level waste : The AVM - Atelier de vitrification de Marcoule-which started operation in June 1978 has vitrified on 1/3/86 1033 cubic meters of fission products representing 190 million curies now stored in 1350 canisters of glass.
Characterisation of waste package is carried out, providing a comprehensive assessment of material properties to comply with basic safety regulations stipulated by french authorities and with agreements for long-term storage by ANDRA and to provide basic data for repository design work. Paper [28] deals with high level waste glass package caracterisation.

The radioactive waste production and delivery to ANDRA which can be forcasted from the french nuclear program is presented in paper [29] :

- Category A or beta-gamma wastes (low and medium activity, short-lived for less than 30 years), coming mainly from everyday running of reactors and fuel cycle and from users of radioisotopes (10%). They will total 800 000m3 in the year 2000.
- Category B or alpha-bearing wastes (low and medium activity, containing a significant amount of long-lived radioisotopes, in particular alpha emitters), stemming mainly from the fuel cycle and in particular, from reprocessing. 75 000 m3 will be produced by the year 2000.
- Category C or vitrified wastes (high activity wastes from reprocessing which are or will be vitrified). By the year 2000 the production will be 3000 m3.

Only the category A waste is considered suitable for disposal in shallow land repository : The Manche Center is in operation since 15 years ; a new disposal center will be opened in 1990.
Categories B and C are kept in engineered storage before final disposal for which the main development and realisation steps are :

- choice of a site for an underground geological laboratory in 1988.
- beginning of construction of a deep repository for category B waste around 1995.
- opening of a deep repository for category C waste around 2010.

7 - REPROCESSING AND WASTES RESEARCH AND DEVELOPMENT

It is not necessary to recall the main stages of the PUREX process. But, it is interesting to recall the specific characteristics of reprocessing and wastes technologies, as described in paper [2] :

- Each different large component is manufactured in very small number, not in series like for reactors or enrichment plant, and most of them can be changed or modified, even though this is difficult, during the life of the plant;

- the process involves a succession of distinct mechanical or chemical engineering operations requiring separate premises and large capacity areas.
- the radioactivity handled (in the order of 1 billion curies and 10 tons of Pu a year for a plant such as UP3, for example) means that there has to be radiation protection for virtually all operations with successive containment barriers, redundant systems for preventing the release of radioactivity and remote control systems (with increasing use of robotics);
- the presence of high concentrations of fissile material in solution means that vessels of very special geometry have to be constructed to take account of the risk of criticality;
- the main incidents disturbing the functioning of the plant can in the vast majority of cases be reduced to four causes:
  . mechanical equipment failures in a hostile environment which makes remedial action difficult;
  . equipment put out of service through corrosion (in the dissolver or evaporator, for example);
  . blocking of pipework by precipitates resulting from the action on solvents of numerous high-radiation bodies, forming complex compounds.
  . malfunctioning of dissolution, extraction or concentration equipments due to inopportune reactions.

These special features, of which this is obviously not an exhaustive list, are enough to show that reprocessing is a difficult and unique activity involving many different trades, numerous specialities and fields of activity in which R & D plays a big part and using the most advanced technologies (robotics, remote control, computer techniques, etc..). Consequently the budgets allocated in France to R&D for the future La Hague and Marcoule reprocessing facilities increased fivefold between 1975 and 1981 (fig.10).

At present, all R & D concerned with reprocessing is based on the PUREX process. Experience has shown that this process is satisfactory for reprocessing oxide fuels from both light water and fast breeder reactors, but much research is underway on improvements. A description of all the avenues being researched would be too long but the main objectives pursued are :
(1) the design of plants with reliable equipment and remote control repair and maintenance facilities giving large reprocessing capacity;
(2) reduction in the volume of waste and effluent;
(3) enhanced safety for installations and for control of fissile materials;
(4) development of high-safety techniques for waste containment.
Work on these objectives is proceeding under three headings : the process itself, equipments technology and waste processing as described in paper [2].

CEA R & D means are described in papers [11] and [12] :
- active chemistry and chemical engineering laboratories in Fontenay-aux-Roses,
- industrial prototypes testing and pilote plant in Marcoule;
- analytical methods and corrosion laboratories in Fontenay-aux-Roses, Saclay and Grenoble;
- low and medium level wastes R & D in Cadarache;
- high level wastes R & D in Marcoule;
- process books working out in Fontenay-aux-Roses and Marcoule.

The R & D program for UP3 and UP2 800 described in papers [11] and [15] includes in particular the following equipments :
- shear of fuel bundle with horizontal feed with emphasis on the essential components: blade, rollers and rails, internal ventilation of the chopper, junctions with chemical equipments.
- continuous rotary dissolver : a major effort has been devoted to the development of continuous dissolution both for process and technology, as illustrated in paper [16].

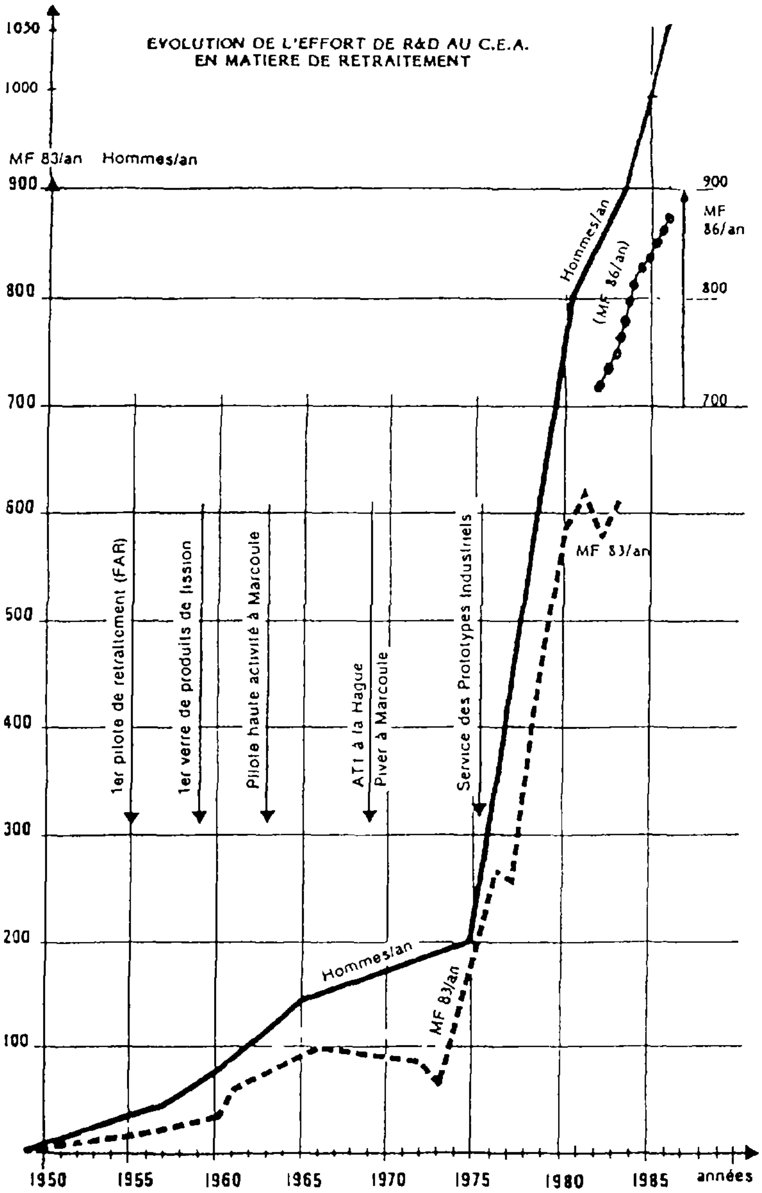


FIGURE 10

- DPC 900 centrifugal pendular decanter
- extraction modelisation
- geometrically safe pulsed columns
- solvent recycling by TBP-diluent vacuum distillation
- PuO_2 redissolution
- iodine 129 trapping in
- vitrification of fission products concentrates by the continuous process experienced in AVM Marcoule facility and adopted at a larger scale for R7 and T7 La Hague facility and also for the UK Sellafield facility as explained in paper [17].
- bituminizing of coprecipitation sluges, evaporation concentrates and ion exchange resins.

Maintenance options adopted for the UP3 plant in relation with the above equipments are described in paper [18].
Construction materials for reprocessing plants are discussed in paper [30].

The R & D program for FBR reprocessing (TOR and MAR 600 Project), described in papers [13] and [19], covers in particular :
- opening of the hexagonal wrapper, pin bundle extraction and removal of the pins.
- pins chopping
- continuous helicoïdal dissolver
- feed clarification
- solvent extraction
- large capacity PuO_2 préparation (oxalate) and packaging systems described in paper [20]
- vitrification
- hulls conditionning by melting in induction autocrucible illustrated in paper [21].

The french R&D waste management program is presented in paper [27].

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LWR SPENT FUEL MANAGEMENT IN THE FEDERAL REPUBLIC OF GERMANY

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Abstract

The spent fuel management strategy in the Federal Republic of Germany is based on interim storage and subsequent reprocessing of spent fuel. The waste will be disposed of in a geologic repository. In parallel, alternative back-end fuel cycle techniques such as the direct disposal of spent fuel without reprocessing are being investigated and might at a later date be used for the final disposal of spent fuel, which is not suitable for reprocessing.

An outline of the national programme on storage, reprocessing and transport of spent fuel and the development and operation of casks for storage and transport are presented.

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1. Introduction

The back-end of fuel cycle strategy in the FRG is based on interim storage and subsequently reprocessing of spent fuel. The waste shall be disposed of in a geologic repository. Parallel, alternative back-end fuel cycle techniques such as the direct disposal of the spent fuel without reprocessing are being investigated and might at a later stage be used for the final disposal of spent fuel, which is not suitable for reprocessing.

Based on the estimated accumulation of spent fuel by the end of this century the German back-end of fuel cycle requirements will be met in two ways, i.e. reprocessing in a German reprocessing plant with an annual capacity of 350 MTU, for which the licensing procedure is under way since 1982 and the first construction license was issued in September 1985, and reprocessing at the French and British facilities of COGEMA and BNFL. Therefore, temporary storage capacity of nuclear power plant pools and additional away-from-reactor (AFR) storage facilities plays an important role in spent fuel management in the FRG besides of reprocessing.

In general, nuclear energy in the Federal Republic of Germany continues to progress towards a normal, fully established and more and more generally accepted energy source. It is also characterized by the fact that industry is taking over full responsibility in the LWR fuel cycle.

2. Interim Storage of Spent Fuel

2.1 Wet Storage at Reactor (AR)

The storage of spent fuel is an interim step in the nuclear fuel management of power reactors. In earlier years, AR storage was considered to be limited to one or two years, allowing for the fuel to cool before being transported to the reprocessing facility. For various reasons the time spent in storage has been increased. Modifications to storage plans had and still have to be made available to prevent shortfalls in storage capabilities at the reactor.

For water pool storage at Reactor - as in commercial use in the FRG - the fuel ponds in newer reactors are equipped with compact spent fuel storage racks. Also in the older plants some pools are equipped with compact storage racks. The following table gives an overlook on the extent of the use of compact storage racks in the fuel storage ponds of BWR and PWR reactors in the FRG.

Compact Storage Racks (on Order) References in Germany

Plant	Name Plant	Type	Power (MWe)	Storage Capacity Positions
KKU	Unterweser	PWR	1300	615 in operation
KKG/BAG	Grafenrheinfeld	PWR	1300	715 in operation
GKN 1	Neckarwestheim 1	PWR	800	486 in preparation
KWG	Grohnde	PWR	1300	768 in operation
KBR	Brokdorf	PWR	1300	768 assembled
KKI 2	Ohu	PWr	1300	768 in preparation
KKe	Emsland	PWR	1300	768 in preparation
GKN 2	Neckarwestheim 2	PWR	1300	768 in preparation
KWO	Obrigheim	PWR	340	980 in preparation
KKK	Krummel	BWR	1316	1582 in operation
KRB-B	Gundremmingen B	BWR	1300	3210 in operation
KRB-C	Gundremmingen C	BWR	1300	3210 in operation
KWB-A	Biblis A	PWR	1200	588 in operation, license is limited
KWB-B	Biblis B	PWR	1300	584 in operation, license is limited
Mulheim		PWR	1300	693 assembled

There is a strong basis of experience that water storage in standard or compact racks of spent LWR-fuel is a mature, viable technology without technological difficulties. There is a substantial basis to conclude from the available experience that Zry clad fuel has not degraded so far. As an example for the given conclusion the results from a demonstration of wet storage performance is given.

Spent fuel rods - intact and operational defective rods - were included in the storage test program. Within 7 years the spent fuel rods were inspected four times. To characterize the spent fuel rods the following methods were applied during pool inspections:

- visual inspection
- profilometry
- eddy current testing
- oxide thickness recording.

Summarizing the results of the intermediate and of the final inspections it has to be concluded that - as predicted - no change exceeding the detection limit could be found neither at the intact nor at the operational defective fuel rods.

2.2 Dry Storage AFR

2.2.1 Dry Storage Facilities in the FRG

Dry interim storage in transport/storage casks is a licensed technology in the FRG. Several types of the CASTOR casks for both storage and transportation have already been licensed. Additional licenses for casks including e.g. the TN 1300 cask are expected in the near future.

The actual status of licensing and construction of AFR interim storage facilities at different sites in the FRG can be described as follows:

- Construction of the first 1500 MTU AFR-facility at Gorleben was started in February 1982. Construction of the facility is completed, the operating license has been issued in 1985. At the time being the license is contested in court.

- License application for a second 1500 MTU facility was filed in October 1979 for the Ahaus site. The construction license was issued in October 1983. Site construction work has started in July 1984. However the construction license is contested in court.

- In combination with the German reprocessing plant at Wackersdorf license was applied in 1982 for a 1500 MTU facility. The first construction license was issued in September 1985.

2.2.2 Description of Storage Technology

In the FRG cast nodular iron was first used for the construction of large transport-storage casks. The first development program and licensing procedure was initiated as early as 1979. Engineering and development was executed by GNS (Gesellschaft fur Nuklear-Service mbH), which obtained the first type B certificate for a CASTOR cask in 1980.

The transport-storage cask technology has found applications not only for storage of spent LWR fuel but was tested also for spent research reactor fuel, spent fast breeder fuel and spent high temperature gas cooled reactor fuel. The casks of the first development phase have relatively limited capacities due to the fact that shortly cooled fuel (ca. 1 year cooling time) has to be accommodated. Advanced models with larger capacities for spent fuel with extended cooling periods are under development. The design for casks of the CASTOR and TN 1300 series is based on type B licensing criteria as established by IAEA. For storage purposes, protections against external and internal events (e.g., aircraft, crashes), the two barrier lid concept with interspace pressure monitoring, sufficient passive decay heat dissipation capabilities and the fuel integrity concept (storage conditions prevent fuel faillure, however activity retention capability concept assumes 100 % fuel rod faillure) constitute the additional design basis.

The casks consist of the following main components:

- The cask body, which is one large piece of cast nodular iron, which incorporates the integrated neutron shielding.
- The closure system, consisting of two lids with corresponding sets of high quality seals, bolts and nuts.
- The basket to hold the fuel assemblies, made of borated stainless steel (CASTOR) or aluminium (TN 1300).
- The lifting trunnions.

The cavity will be filled with inert gas (Helium) and the space between the lids form a pressure barrier which can be monitored continuously during storage by a pressure gauge.

The storage facility at Gorleben is basically a large storage building confirming the casks standing in upright positions. The decay heat will be removed from the cask surfaces by natural convection which is facilitated by openings in the side walls and the roof of the building. A total of 420 positions is available providing a capacity of 1500 MTU.

2.2.3 Summary of R+D-Work

2.2.3.1 Spent Fuel Storage Performance

R+D-work was carried out, to assess the integrity of spent fuel assemblies and their handling capability throughout dry storage periods over a decade and more, during the discharge of the fuel assemblies from the storage locations and during shipment to fuel reprocessing. The research programs covered theoretical predictions, laboratory work and performance testing of real spent fuel.

The results of the R+D activities in the FRG can be summarized as follows:

- the experimental findings and the theoretical analysis are in agreement

- no indications of ISCC or crack growth in the Zircaloy cladding
- tangential strain and cladding oxidation are in the range of the detectability limit under FRG cask storage conditions
- moisture can be removed from operational defective rods during the cask drying operation
- no strong contamination of the cask inner surface and basket
- no indication of any further propagation of cladding defects under inert conditions
- 400 °C is a safe and reliable upper temperature limit for dry storage in inert atmosphere. Even somewhat higher temperatures seem to be acceptable.

2.2.3.2 Cask Performance

Dry storage tests and demonstrations of storage cask performance have involved ~ 3000 fuel rods, which have been monitored (Krypton-85) during dry storage with maximum cladding temperature ranging from 250 to 450 °C. There is no evidence that any fuel assemblies loaded into the casks have failed during transportation and storage. All procedures as cask loading, inertisation, decontamination, transportation and unloading performed well within the limits prescript by the licenses.

2.3 Reprocessing

A very important event was the decision to construct the first commercial reprocessing plant. The reprocessing company (DWK), set up by all German utilities operating nuclear power plants, took that decision in February 1985 in order to meet the requirements of spent fuel management of commercial nuclear power plants in the Federal Republic. The new industrial complex to be built at Wackersdorf in Bavaria will also comprise a fuel reception and storage facility, a fuel fabrication plant, a radioactive waste conditioning plant and a storage facility for conditioned waste. The beginning of cold operation of the plant is scheduled for 1993. The total investment cost is estimated at 5 billion DM, to be financed exclusively by private funds.

The Federal Government sponsored an assessment of direct disposal of spent fuel versus reprocessing. It was based on a comprehensive study of the safety of direct disposal, conducted by the Karlsruhe Nuclear Reserach Center. The study came to the conclusion that direct disposal does not offer decisive safety advantages and still falls short of the level of technical maturity already established for reprocessing.

The construction of a pilot scale reprocessing plant was completed in the FRG already in 1971 by commissioning the WAW in the Nuclear Research Center at Karlsruhe. The capacity of the WAK is 30 - 40 t/a. The operation license allows for the reprocessing of fuel assemblies with an initial enrichment of 3,2 % U-235 and a maximum burn-up of 40 GWd/tU. In 11 campaigns 120 t of LWR-fuel were reprocessed successfully. The experiences gained provides the basis for the design, construction and operation of the WAW together with an additional R+D-program. The WA at Wackersdorf with its capacity of 350 t/a is a scale up of about 10:1.

2.4 Thermal Recycling of Pu and U

The Federal Republic of Germany (FRG) decided to close the fuel cycle by erecting the reprocessing plant WA350 at Wackersdorf. As long as the plutonium supply from reprocessing plants exceeds the plutonium of fast breeder reactors, recycling of plutonium in LWRs is a convenient solution by which a significant advanced uranium utilization is achieved.

The German utilities decided in 1980 on a program for the thermal recycling of all the excess plutonium which is not ready for immediate use in breeder reactors, and to cooperate with each other and together with KWU/ALKEM. The objective is to increase sufficiently the MOX fabrication capacity for the demonstration of the large-scale technical feasibility and the economic use of plutonium in LWRs.

The demonstration of plutonium recycling performed to date in the FRG in LWRs shows that thermal plutonium recycling on an industrial scale is feasible and that the usual levels of reliability and safety can be achieved in reactor operation.

2.5 Final Disposal of Radioactive Wastes

High level waste solutions are generated by the reprocessing. The FRG-concept forees to solidify those wastes after a suitable decay period by vitrification. The PAMELA demonstration plant for vitrification at MOL/Belgium has started its test program. The exploration of the salt dome at Gorleben in its geological vincinity is completed. The construction of the 2 shafts for further exploration of the interior of the Gorleben salt dome in order to check its suitibility as a geologic repository is continued according to schedule.

The KONRAD iron ore mine, presently converted into a repository for low level wastes will become operational by 1989.

2.6 Direct Disposal of Spent Fuel Assemblies

Spent fuel disposal without reprocessing has been under investigation in the FRG since 1979. Up to now, technical concepts under investigation for this waste management route have been developed, and technical feasibility has been demonstrated in principle. Conceptual engineering has been carried out for a conditioning and encapsulation plant enabling the encapsulation of all kinds of spent nuclear fuel. This plant shall be built in Lower Saxony, and the licensing procedure is expected to start in 1986. In a first step this plant will be used to demonstrate different conditioning and encapsulation techniques with a maximum annual throughput of 35 t of spent fuel. According to the present time schedule, the cold and hot operation could start in 1992 and 1994 respectively.

As far as LWR spent fuel is concerned, two types of spent fuel packages can be manufactured in principle:

- intact spent fuel rods, removed from the fuel assemblies encapsulated in heavily shielded disposal canisters which are emplaced in the disposal tunnels of a repository.

- Alternatively, disassembled fuel pins can be cut into pieces of about 1 m length and encapsulated in canisters which have the same outer dimensions as the canisters for vitrified reprocessing waste.

A demonstration and test program on a 1:1 scale is under preparation to demonstrate the safe and reliable handling of both types of canisters in a repository and to investigate special thermal and rock mechanics aspects of the tunnel emplacement concept. Most experiments are performed in the ASSE salt mine, the German R+D Underground Laboratory.

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CURRENT STATUS OF SPENT FUEL MANAGEMENT IN JAPAN

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Abstract

Japan now operates 33 nuclear power reactors with a combined capacity of 24,686 MWe. The future nuclear power generation capacity is forecasted to be about 34 GWe in 1990 and 62 GWe in 2000. The basic nuclear spent fuel management concept is based upon the promotion of fast breeder reactors. Plutonium obtained from spent fuel through reprocessing will be used in FBRs, in advanced thermal reactors and also in LWRs. The reprocessing of spent fuel of LWRs is the chosen spent fuel management scheme selected by Japan.

The state of the art and prospects of spent fuel storage, reprocessing in the country and outside, and experience on sea transportation of spent fuel are described.

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1. Introduction

Since Japan is mostly dependent on foreigen countries in its energy resources, the diversification of energy sources is a very important problem from the view points of the energy security. The usage of the nuclear power is the key option for the future energy source because the recovered uranium and plutonium can be fed to the nuclear reactors. Thus the establishment of the nuclear fuel cycle is one of the basic nuclear policies.

Nearly twenty years has passed since Japan's first nuclear power reactor began commercial operation in July, 1966. Japan now operates 33 nuclear power reactors with a combined capacity of 24,686 MW and ranks fourth in the world. The future nuclear power generation capacity is forecasted to be about 34 GW in 1990 and 62 GW in 2000, respectively.

The basic nuclear power development policy of Japan lies in the promotion of fast breeder reactors (FBR) to follow light water reactors (LWR). In working out this basic policy, the use of the plutonium obtained from spent fuel through reprocessing, not only in FBRs and advanced thermal reactors (ATR), but also in LWRs, is corsidered to be an important option for future nuclear power development. Japan is conducting research and development in the related areas.

The reprocessing of spent fuel of LWRs is, therefore, an operation that must be promoted in Japan.

2. Basic policy for spent fuel management

Plutonium and uranium recovered from spent fuel can be utilized as domestic energy resources and their uses will lessen the dependence of nuclear energy resource on foreign country.

The annual amount of spent fuels in Japan is predicted to be about 800 tons in 1990 and 1,500 tons in the year 2000, in view of the scale of nuclear power generation in the future. Reprocessing of spent fuels is important not only in the promotion of plutonium utilization, but also in the proper management and disposal of the radioactive wastes from the spent fuels. At present, the reprocessing is mostly entrusted to overseas reprocessing facilities. However, on the principle of domestic reprocessing, a private reprocessing facility must be constructed in addition to the Tokai Reprocessing Plant operated by Power Reactor and Nuclear Fuel Development Corporation (PNC), to meet the demand for treatment of spent fuels in the future.

The reprocessing plant with a capacity of 800 tons a year has been promoted by the private sector and will begin the operation around 1995.

The Government supports the research and development activities, such as demonstration of technology for major components, improvement of safety and reliability of the facility, reduction of the amount of radioactivity released into environment, and improvement of reliability of safeguards.

The Government also supports the plan so as to permit smooth progress in securing the site for this plant, and in the field of fund raising.

Based on the basic policy of reprocessing of spent fuel, it is not expected to manage the storage of spent fuels for a long period of time, except for tentative storages at reactor sites (AR) or at reprocessing plant sites. Therefore, there is no concrete plan about long-term storages at reactor sites or at away from reactor (AFR) sites.

However, since the timing and the amount of reprocessing should be decided flexibly considering the form of utilization of recovered plutonium and recovered uranium (i.e. usage, material balance, etc), the storage duration and the amount of the spent fuels tentatively stored at reactor sites or at reprocessing plant sites will also be decided flexibly.

In addition to the storage in water pool, the studies of dry storage technology are conducted continuously both by the government and private companies in order to meet the future situations within the framework of the principle of spent fuel management.

3. Present status and perspectives

3.1 Storage of spent fuels at nuclear power plants

About 1,600 tons of LWR spent fuels are stored in the pools at reactor sites at the end of 1985.

Accumulated amount of LWR spent fuels stored in the pools would be about 2,800 tons at 1990, 3,600 tons at 1995, and 7,000 tons at 2000, under the condition that spent fuel is transported to the Tokai plant and to overseas reprocessing facilities, and if the second reprocessing plant could not start the operation until 2000.

The storage densification technology has been applied to the existing pools to enhance the utilization, and newly built reactor has larger storage capacity than older one.

3.2 Transportation of spent fuels

3.2.1 Domestic transportation

(1) Present status

Spent fuels from LWRs are shipped to the Tokai reprocessing plant by sea route. Sea transportation is the most efficient way in Japan because both power stations and reprocessing plant are located at the sea coast. Five HZ-75T casks, which are owned by a specialized private firm and are leased to electric utilities, are currently in operation. A ship (Hinoura-maru) of nominal 1,200 d.w.t. being operated by a private firm can carry four HZ-75T casks. (Fig. 1, Table 1)

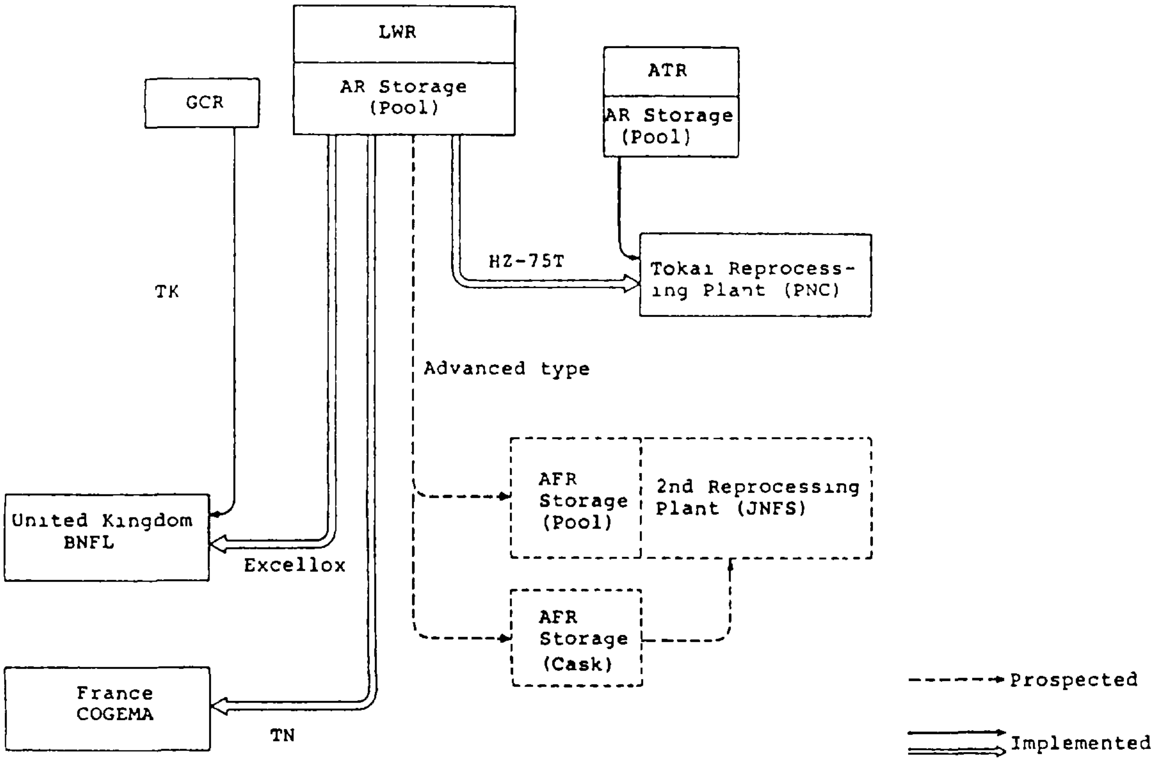

Fig. 1 Schematic Flow of Spent Fuels in Japan

Table 1 Main Features of Casks

Item \ Cask		HZ-75T	EXL-3B	EXL-4	TN-12A	TN-17
Type		B(M)/Wet	B(M)/Wet	B(M)/Wet	B(M)/Dry	B(M)/Dry
Overall Dimension (m) (incl. shock absorber)		5.9L x 2.3ø	6.0L x 2 1ø	6.3L x 2.4ø	6.2L x 2.5ø	6.2L x 2.0ø
Payload Number of Assemblies	BWR	17	-	-	32	17
	PWR	7	5	7	12	
Package Weight (Approx. max., t)		80	70	97	102	79

Note 1; TK Model
Overall dimension (m) Approx. 2.6L x 1 8W x 2.0H
Package weight Approx. 46 ton
Payload, No. of assembly 260 (3.28 ton)

Note 2; NH-25
B(M) type
Overall dimension (m) 5.8L x 1.5ø
Payload, No. of assembly BWR 2, PWR 1
Package weight Approx 29 ton

Major electric utilities and PNC have set up cask control centers at their reactor sites and plant site for the efficient use of casks and ships.

Tokyo Electric Power Co. has, for example, such a center at Fukushima site, where maximum 26 casks can be handled at a time.

Besides usual shipments a special cask NH-25 is used to transport spent fuels for the post irradiation examination to hot laboratories.

(2) Future prospect

At the time of introduction of the second reprocessing plant in the 1990's, much more advanced type of transporting casks for high performance fuel would be required. Electric power companies are promoting the development program in cooperation with industry.

3.2.2 Overseas transportation

To data more than approximately 2,700 tons of spent fuels have been shipped to United Kingdom and France by James Fisher & Sons Ltd., and Pacific Nuclear Transport Ltd. (PNTL). PNTL which was set up in 1975 exclusively for the transportation of spent fuels from Japan to Europe, has an existing fleet of 58 BNFL-designed Excellox type and 34 TNP (Transnucleair Paris) desinged TN type casks, in addition to four transporting motor vessels [Capacity; Pacific Swan (20EXL/14TN), Pacific Crane (22EXL/16TN), Pacific Teal (24EXL/16TN) and Sandpier (20EXL/15TN+8PK)] of nominal 3,000 d.w.t. class in operation.

Some specific features of casks are given in Table 1. The mixture of casks reflects the requirements of Sellafield and La Hague facilities. Magonx fuels from GCR has been exclusively shipped to the United Kingdom by CEGB-designed TK casks.

3.3 Reprocessing

3.3.1 Tokai plant

Since the Tokai plant, the first reprocessing plant in Japan, started the hot operation in September 1977, 299 tons of LWR fuels have been received and 253 tons have been reprocessed by the end of 1985. The plant is owned and operated by PNC.

Through the plant operation experienced so far, the performance has been satisfactory for such processes as extraction, separation, purification, and treatment of low-level liquid waste. High availability of about 90 percent on the average has been recorded for the extraction and separation processes during feeding the dissolved solution continuously. The accountability control has also been successful.

During the hot test operation of the plant a leakage of radioactivity from the acid recovery evaporator was detected in August 1978, followed by the replacement of the evaporator to a new one. For this occurrence the plant was shut down for 14 months for troubleshooting, dismantling of the evaporator and manufacturing and installation of a new one.

The plant was commissioned in January 1981. In April 1982 a leakage of radioactivity into steam condensate used for heating a dissolver was found and the steam line was disconnected from the other process lines. Afterwards the plant operation was continued using another dissolver. In February 1983 the second dissolver experienced a similar throuble as with the first one, and also the acid recovery evaporator showed again a leakage of radioactivity into the steam line. The plant was shut down again for repair.

The remote repairing technology that had been developed for the first defective dissolver could be applied to the second one, and further it was decided to install additionally a new dissolver made of high cromium and nickel content austenite series stainless steel with improved design features.

Remote repair work for the two existing dissolvers was successfully completed and the test reprocessing was done in December 1983 using two tons of spent fuel.

Decontamination work for the installation of the new dissolver began in March 1983 and the installation work was completed in March 1985 including the test operation of the new dissolver using five tons of spent fuel. The plant operation with three dissolvers was started in April 1985, and 74 tons of spent fuel have been reprocessed by the end of 1985.

It is expected that the plant will process LWR spent fuels at around 90 tons per year on the average from 1986 to 2000. Plant shutdown for heavy maintenance as long as one year will be intervened at least twice during the 15 years operation.

3.3.2 Plan of commercial reprocessing plant to be constructed by JNFS

Japan Nuclear Fuel Sercice Co., Ltd. (JNFS) was established in March, 1980 with the capital investment of the leading one hundred companies in related industries in Japan, such as electric power, steel machinery, chemical, and finance.

The Company is to construct and operate plants for reprocessing spent fuel received from nuclear power plants in Japan, and to supply the power industry with the recovered nuclear fuel materials.

In July, 1984, the Federation of Electric Power Companies proposed plans to site the nuclear fuel cycle facilities at Rokkasho-mura, Aomori Prefecture, and the local authorities dicided to accept the siting proposal in April, 1985. For the construction of the plant, surveys similar to those for the construction of the nuclear power plant are under way with completion scheduled for some time toward the end of 1986.

Work on the design of each facility for the plant was started in 1984. On the main reprocessing facilities, in particular, Design Criteria Study have been carried out in preparation for basic design.

The main operations of the Company are as follows;

(1) Storage of spent fuel

The spent fuel transported from nuclear power plants all over Japan will be stored in pools at the plant. Port facilities will be provided at Mutsu-Ogawara Port where harbour improvement work has been implemented, and storage facilities consisting of pools and related installations with a initial capacity of 3,000 tons of spent fuel will be constructed.

(2) Reprocessing of spent fuel

The spent fuel cooled in the storage pools for the required cooling time will be chemically processed to have the residual uranium, the generated plutonium, and the fission products separated through solvent extraction. For these processes, the main reprocessing plant with an annual throughput of 800 tons will be constructed, with the commissioning scheduled for some time around 1995.

(3) Purification and storage of recover ed fuel materials

The separately extracted uranium and plutonium will be purified, and then converted into uranium oxide and plutonium oxide, respectively. For the temporary storage of these oxides before shipment, the construction of storage and related facilities will be completed at some time around 1995.

(4) Investment costs

The total construction costs of the plant are estimated at about 700 billion Yen in terms of 1984 price levels.

The fund will in principle be raised through capital investment, relying mostly on borrowing, primarily from the Japan Development Bank.

3.3.3 Reprocessing in foreign countries

The Electric Power Companies have made reprocessing contracts with BNFplc and COGEMA by which about 4,800 tons of LWR spant fuels would be reprocessed. Japan Atomic Power Co. has made reprocessing contract with BNFplc in which about 1,100 tons of GCR spent fuels would be reprocessed.

Under these contracts spent fuels discharged from nuclear power reactors in Japan have been transferred to the United Kingdom and France.

4. R&D activities supporting spent fuel management

4.1 LWR fuel reprocessing

4.1.1 R&D activities by PNC

The availability of Tokai plant was 34 and 25 percent in 1981 and in 1982, respectively, and the operation was interrupted in 1983 because of the leakage occurred at dissolvers. Major factors affecting the plant availability were defects on integrity of such components as dissolvers and evaporators, mechanical process failures, inefficient filtering of dissolved solution, clogging of transfer lines for high-active solution, and other troubles with components having no redundancy.

Based on these experiences it was recognized that supporting R&D activities be strengthened to accomplish more stable operation of the plant, therefore, the R&D programs were reviewed and revised in 1983.

Two kinds of R&D programs are carried into effect; one the short- and medium-term program to improve plant operability and the other the long-term program to attain extended stable operation.

The short- and medium-term program includes such items as process control system improvement, development of components with higher reliability, development of remote repair and in-service inspection technology, etc. In the long-term program emphasis is placed on the advanced engineering for head-end processes including mechanical chopping of fuel elements, dissolution and dissolver off-gas treatment, and clarification. Improvement of process performance is the main objective of this program to be accomplished by engineering testing of advanced equipments, proper layout of components, quality assurance through the development and demonstration of higher plant availability. Major items of R&D programs for reprocessing plant technology being undertaken at PNC are listed in Table 2.

For the development of the second reprocessing plant in Japan PNC is going to have several cooperative R&D programs with JNFS. The first one is the verification of chemical process parameters for the plant.

4.1.2 R&D activities for large-scale reprocessing plant.

With the cooperation and assistance of the government and electric power companies, JNFS has been promoting the development of technologies for application to larger scale confirmation, environmental safety and other purposes.

Most of these technology development efforts intended for environmental safety are due for completion in fiscal 1984, and for larger scale condirmation, in fiscal 1986.

Table 2 Major items of R & D programs for reprocessing plant technology being undertaken at PNC

Item	Period	Present status
o Maintenance technology		
• remote reparing of dissolver	1982 ∿ 83	remote welding machine developed and used successfully
• remote decontamination of dissolver loading cell	1983 ∿ 84	decontamination machine for dissolver loading cell developed and used successfully
• remote dismantling of dissolver	1984 ∿ 87	under design study
• remote maintenance of HLW piping system	1984 ∿ 88	under design study
o In-service inspection technology	1984 ∿ 88	remote inspection equipments being developed
o Development of highly reliable components		
• acid recovery evaporator	1984 ∿ 88	under design
• plutonium solution evaporator	1984 ∿ 88	under design
o Process control system improvement	1980 ∿ 88	under system design for computer control
o Improvement of instrumentation and analytical methods	1976 ∿ 88	alpha monitor, phase-interface detector etc., being developed improved criticality detection and alarm system completed
o Advanced engineering development for head-end processes		
• mechanical chopping process	1984 ∿ 92	under design study
• dissolution and dissolver off-gas treatment processes	1984 ∿ 92	under design study
• clarification process	1984 ∿ 92	under design study
• engineering testing and demonstration	1986 ∿	under construction of a new engineering demonstration facility
o Waste management technology		
• bituminization of low-level liquid waste		hot test operation being continued
• plastic solidification of used solvent		hot test operation being continued
• krypton recovery (cryogenic process)		test facility completed and cold test being continued
• vitrification of high-level liquid waste		hot process test being continued and a pilot facility being desinged

4.1.3 R&D in JAERI

R&D activities in Japan Atomic Energy Research Institute (JAERI) are categorized into three phases, i.e., the first in the 1960's of design and construction of the first pilot facility by the aqueous method in Japan, the second phase in the 1970's of several process evaluation studies on some alternative processes to the aqueous reprocessing and the third phase at present of researches focussed on the current Purex process.

In the pilot facility project in the first phase, many process-chemical and enginering-scale studies had been conducted for the design, and not campaigns with about 200 kg-heavy metal per batch hand been successfully performed in the cooperation with PNC.

In the studies on non-aqueous reprocessing in the second phase, through bench- or engineering-scale experiments were evaluated such alternatives as the fluoride volatility process for FBR fuel, an improved Voloxidation process for the tritium retention from LWR fuel and an advanced head-end process for HTR fuel.

In the 1980's when the program of the commercial reprocesing plant have been started, R&D activities in JAERI are being reoriented toward the aqueous reprocessing. These activites cover such research areas as safety research, advanced process development including related process chemistry and development of improved safeguards technologies.

In the safety research, emphasis is placed on such items as criticality safety, radiation shielding, confinement performances of ventilation systems and safety evaluation in abnormal or accident conditions. These include computer code development, bench-scale experiments and engineering system demonstration.

The process development research aims at improving the current Purex flowsheet by applying advanced processes, with the view of the adaptation to high burn-up fuels, environment safety, simplification of the reprocessing waste management. They consist of many fundamental experiments and some bench-scale experiments.

Table 3 Major researches presently conducted or planned in JAERI on fuel reprocessing

Research item	Objected research level[a)]	Started in the FY of	Present status
1 Safety Research for Nuclear Fuel Cycle			
·Criticality Safety (nuclear phenomena, process chemical behavior)*	B(H)-E(H)	1980	Preliminary experiment continued.
·Filter System Test (severe condition and posturated accident conditions.)	E(C)	1981	Finished for fire accident conditions
·Basic data for safety evaluation (accident source-term*, Zircalloy corrosion in dry storage, radiolysis of solvent and water)	B(H)	1985	Preliminary experiment stated.
·Safety evaluation methodologies (accident evaluation code, process simulation code*)	A	1984	System analysis continued
2 Process Development			
·Dissolution test of high burn-up fuels*	L(H)	1983	Hot experiment started.
·Advanced reprocessing flowsheet*	L(C)-L(H)	1985	Cold study started.
·Modified partitioning process of HLW*	L(H)	1977	Primary flowsheet demonstrated.
·Process for environmental safety improvement (Iodine, Tritium)*	B(C)-L(H)	1982	Cold study continued.
·Advanced process for TRU waste treatment (spent solvent, ML aqueous waste, solid wastes*)	L(C) -B(H)	1980 1985	Apparatus completed for ML aqueous waste Scouting experiment started for spent solvent
·Measurement technique for solidified TRU wastes	B(H)	1984	Equipment construction continued.
·Related researches (process chemistry on TRU, FPs and materials research)	L(C)-L(H)	-	Surveys and experiments continued.
3 Research for safeguards			
·Development of advanced materials accountancy system	A	1978	System development continued. Field test of primary system finished.
·Development of containment and surveillance technolgy	E(-)	1981	Field test continued.

a) Symbolic characters indicate objected research levels with radioactivity used as follows,

A : Analytical research including code development, L : Laboratory-scale experiment,
B : Bench-scale expriment, E : Engineering-scale experiment, (C) : Cold experiment with U and/or tracer,
(H) : Hot experiment

b) Research items with asterisks are planned to be conducted in the NUCEF program. (See tent.)

In the research on the safeguards technology, are continued design and evaluation studies of effective safeguards systems, and development of elemental technologies for their improvement and optimization. The research on non-destructive assay techniques to measure the amounts of heavy elements in spent fuels is also included.

Major research items are summarized in the Table 3 with their present status. Some of them are conducted in close cooperation with PNC or JNFS. In order to advance these researches more efficiently and extensively, integration of the related programs in JAERI is in progress as well as the construction of a new research facility, Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF), that enable R&D with highly irradiated fuel and sufficient amount of fissile materials. The facility is scheduled to be in hot operation around 1990 for experiments on nuclear criticality safety, advanced reprocessing process and transuranium waste management.

4.2 FBR fuel reprocessing

Development of FBR fuel reprocessing technology is a logical extension of fast reactor development. For efficient development of the technology R&D programs are carried out in close relation to the development of LWR fuel reprocessing technology.

The R&D studies started about ten years ago, and since 1982, spent fuel from the experimental fast breeder reactor JOYO has been used for the process study at the Chemical Processing Facility, PNC. The design of a pilot plant having a throughput of 120 kg per day is in progress. The plant is planned to be in full test operation in the middle of the 1990's using fuel assemblies discharged from the prototype breeder reactor MONJU.

4.3 Spent fuel storage

R & D on spent fuel storage in Japan were conducted solely on wet storage technology before the 1980's, because Japanese policy of spent fuel management has been to reprocess all of the spent fuel. Recently, however, studies on dry storage are being started in the consideration of future situations that flexibilities in spent fuel storage might be required. R & D on spent fuel storage are categorized to (1) improvements of the existing technology of wet storage, (2) technical evaluation studies, including fundamental experimental research, of dry storage. The purpose of these studies is to provide basic data for the technical judgement of future storage options that allow flexible spent fuel management.

With regards to wet storage, design and safety analysis for enhanced utilization of existing storage capacity have been conducted by PNC and electric utility companies, and an in-situ data collection is made by PNC for proving integrity performances of cladding in the long-term storage.

Systematic researches on dry storage are being started by Central Research Institute of Electric Power Industry (CRIEPI) and JAERI.

In 1983, CRIEPI studied on applicability of spent fuel storage technologies developed in the world on the basis of literature survey. CRIEPI also developed basic conditions for spent fuel storage in Japan, such as regulations of conformance, storage scenario, generic site, characteristics of spent fuel, design criteria of storage facilities, etc.

In 1984, CRIEPI conducted conceptual designs and cost estimations for dry storage methods and water-pool storage methods based on the above-mentioned basic conditions. Then, CRIEPI will conduct comparison study between a cask storage method which is expectedly chosen by the preliminary selection and a water-basin storage method, from view points of technology and economy. The results show that a cask storage method is most promising for spent fuel storage capacity of 500 tons in the site of nudear power station (AR). In 1985, CRIEPI evaluated an applicable storage method for AFR storage in Japan after another comparison study between a vault storage method which will be alternatively chosen by the above-mentioned preliminary selection and a water-basin storage method, again from view points of technology and economy. It was indicated that for a capacity of 3,000 tons, the vault storage method is most economical in AFR.

From 1984, by a consignment from Science and Technology Agency, CRIEPI is also making a study on spent fuel storage of large amount up to 10,000 MTU with safety analysis including temperature calculation of claddings stored in dry casks.

JAERI plans to share in the experimental studies including oxidation behaviors of irradiated fuel rod in dry storage, in order to provide safety evaluation data.

R&D on elemental technologies for typical storage methods are under consideration. Some related data are expected through operational experiences of a small-scale dry storage facility, that was constructed at JAERI, Tokai, in 1982 for storing the natural uranium spent fuel from the research reactor (JRR-3).

4.4 Management of high-level radioactive wastes

High-level radioactive wastes (HLW) produced by the reprocessing are now stored in the form of solution in tanks at the reprocessing plant. These wastes are to be vitrified and stored to cool off their heat for a certain period (30 ~ 50 years) at an engineered facility, where the government is responsible to demonstrate the respective technologies. Then the high-level wastes will be finally disposed of into repository in geological formations under due control of the government.

It is now at the stage of research and development on HLW. PNC is taking a leading role in Japan in the development of technology necessary for treatment and disposal of HLW. HLW are to be solidified into borosilicate glass (vitrification), and then to be diposed of into geologic formation. A plant for vitrification will start operation at the beginning of the 1990's. Their research and development activities for the geological disposal are being coducted under a longterm plan with a view to establishing the technology for disposal as soon as possible after the year 2000. (Fig. 2)

In parallel with the development of the technologies JAERI is developing methodologies for the safety evaluation of disposal, and also promoting researches on alternative technologies such as partitioning and SYNROC solidification.

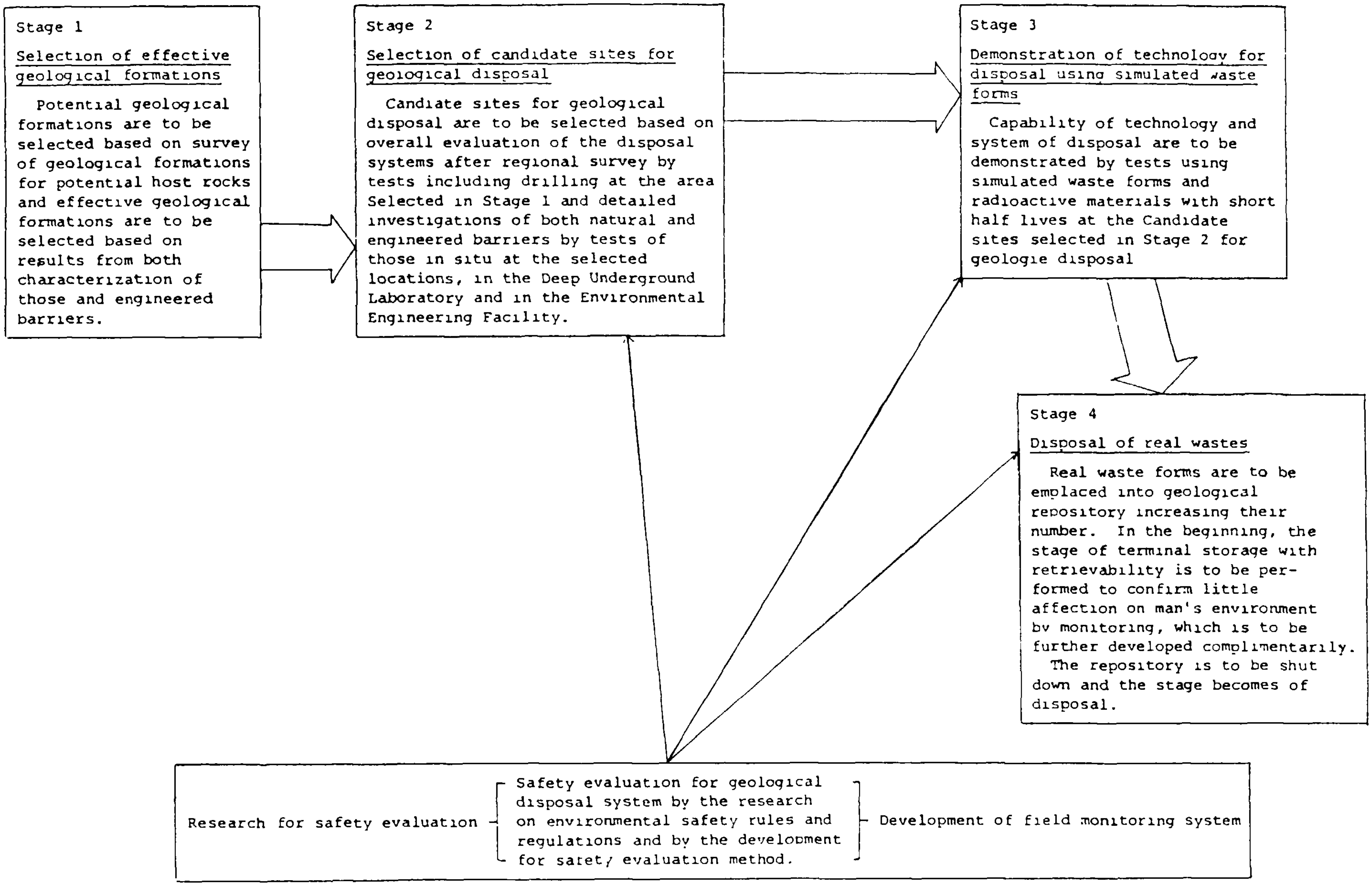

Fig. 2 General view of approach to geological disposal

4.5 Recycling of recovered uranium and plutonium

Recycling the products recovered from reprocessing spent fuels is at a developmental stage. Although plutonium will be fed into FBR eventually, several options or alternatives are considered for the time being.

Plutonium can be used in ATR and/or LWR, therfore, both options are being developed for technical demonstration.

Recovered uranium can either be converted to uranium hexafluoride followed by re-enrichment and re-conversion, or be mixed with plutonium to be recycled as mixed-oxide fuel. All these options are being studied.

CURRENT STATUS OF THE SWEDISH WASTE DISPOSAL PROGRAM

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Presented by B. Gustafsson

Abstract

Direct disposal without reprocessing is the main strategy for management of spent nuclear fuel in Sweden. Under the present circumstances this is the most economic route. Before final disposal the fuel will be stored for about 40 years in CLAB, a specially built underground facility which was put in operation in July, 1985. This will give flexibility in the choice of final disposal method. Wastes from reactor operation will be disposed of in SFR, another underground facility in hard rock with planned operation starting in 1988. The total cost for waste management in Sweden is estimated to 47 billion SEK (5.5 billion US$) or about 0.02 SEK/kWh (2.4 mills/kWh) including all types of waste and also the decommissioning of all twelve nuclear power facilities.

BACKGROUND

The nuclear power program in Sweden consists of 12 nuclear units. Their combined capacity is 9500 MWe. The last two units Forsmark 3 and Oskarshamn 3 reached full power for the first time in May 1985 and are now in their final startup testing phase.

The Swedish legislation explicitly puts the primary responsibility for the management of radioactive wastes from nuclear facilities on the owners of these facilities. This means that the nuclear power utilities in Sweden are responsible for all necessary steps in the handling and final disposal of all radioactive wastes arising from the nuclear power program. The utilities have assigned the duty to perform these steps to their jointly owned Swedish Nuclear Fuel and Waste Management Co (SKB). The responsibilities of SKB in the nuclear waste management field thus include

- all necessary research and development work

- planning and cost calculations for the total nuclear waste management system (except handling and treatment at the reactor sites)

- design, construction and operation of all necessary facilities for storage and disposal of nuclear wastes

- all transportation and handling of spent nuclear fuel outside the reactor sites.

There are mainly three national authorities in Sweden which deal with matters related to radioactive wastes from nuclear power plants:

- The Swedish Nuclear Power Inspectorate (SKI) supervises and controls the safety at design, construction and operation of nuclear facilities;

- The National Institute of Radiation Protection (SSI) supervises and controls that appropriate measures for radiation protection are taken by the plant owner;

- The National Board for Spent Nuclear Fuel (NAK) supervises the planning and the research and development for the waste management program. NAK also administers the funds for future costs for radioactive waste management, which are built up by a fee on the nuclear power production.

WASTE MANAGEMENT STRATEGY

The planning of the Swedish waste management system is based on the operation of the 12 reactors up to the year 2010. The main feature of the planned system are shown in Figure 1, which also gives some data on energy and waste quantities produced to 2010.

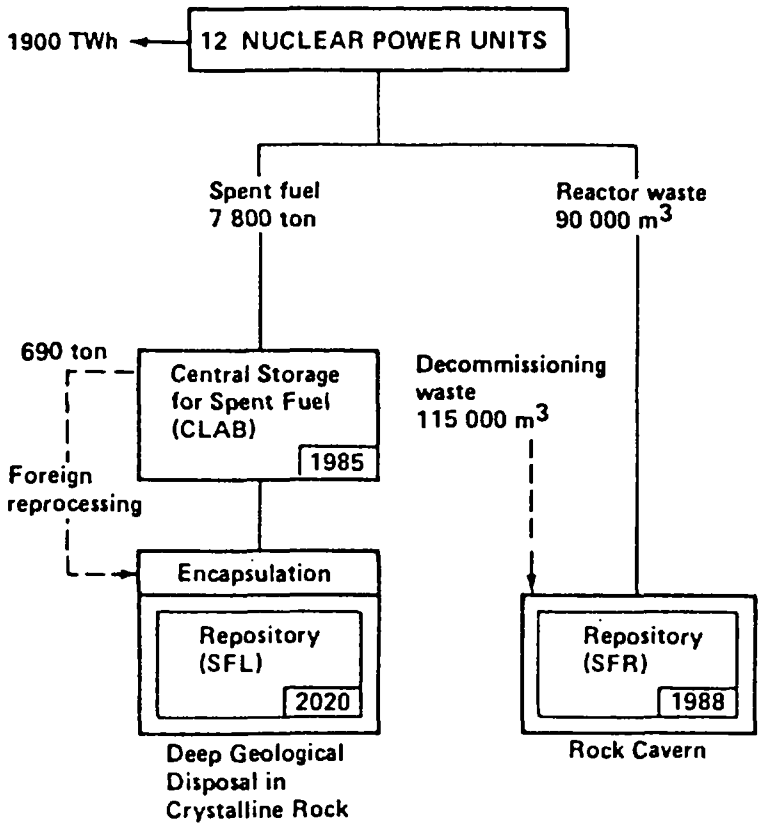

Figure 1. Main system for management of radioactive waste in Sweden

The main strategy for handling of the spent fuel is direct disposal without reprocessing. Under the present circumstances in Sweden this is the most economic way of handling the spent fuel. It is also at present the politically prefered option.

About 690 tonne of spent fuel have however been contracted for reprocessing at Sellafield in UK and La Hague in France. An additional 178 tonne contracted for at La Hague were recently transfered from SKB to a Japanese company. SKB is also actively trying to transfer most of its remaining reprocessing contracts in order to limit the number of different wastes types as much as possible.

The spent fuel will be stored for some 40 years before final disposal according to present plans. This will allow the fuel residual power to decrease considerably and make the final disposal easier. It will also provide great flexibility to adjust to future developments in the area of spent fuel management. Further it will provide ample time for the research and development for the repository site selection and for the system's design and optimization.

After the interim storage period the spent fuel will be encapsulated in corrosion resistantcanisters and shipped to a final repository. The interim storage and the encapsulation facility will give rise to some additional quantities of waste which must be considered in the planning of a complete system. The total amount of conditioned wastes from the Swedish nuclear power programme according to present plans will be about 245000 m^3.

THE INTERIM SPENT FUEL STORAGE FACILITY - CLAB

As mentioned above interim storage of spent fuel plays a strategic role in the waste management system. Construction of CLAB - the central interim storage for spent fuel started in the spring of 1980 and the first spent fuel was shipped to CLAB in July 1985. The facility is located at the site of the Oskarshamn nuclear power plant on the Swedish east coast.

The fuel is stored under water in stainless steel lined concrete pools in a large underground rock cavern with at least 25 m rock cover. There are 4 pools, with a total capacity of 3000 tonne. Additional storage pools can be added when needed up to a total of 9000 tonne. The fuel reception and unloading facilities are in a building above ground. The total receiving capacity is about 300 tonne per year or about 10 spent fuel shipping casks.

With the present plans CLAB would need to be expanded around 1995. It is also forseen that used core components would be stored in a separate pool at CLAB.

TRANSPORTATION OF SPENT FUEL AND RADIOACTIVE WASTE

All nuclear power plants in Sweden are located at the coast and the sea is used as the ultimate heat sink. The interim fuel storage -CLAB- and the final repository for LLW and MLW from reactor operation-SFR- are also on the coast. It has thus been decided to use sea transportation as the main method to ship spent fuel and radioactive waste. This is also quite favorable considering the large transport weights that must be handled with this type of cargo.

A specially built and equipped ship - M/S Sigyn - was taken in operation in 1982. The ship can take 10 fuel casks of the type TN17 Mk2 each with a weight of about 80 tonne. Each cask can take 17 BWR or 7 PWR fuel assemblies. The fuel is transported dry and cooled by natural air convection. The casks are mounted on special transport frames which are handled by a specially built terminal vehicle.

The same equipment can also be used to transport MLW and LLW in containers of steel or concrete. Each container has a transport weight of up to about 120 tonne.

FINAL REPOSITORY FOR REACTOR OPERATIONAL WASTE - SFR

The first repository to be constructed in Sweden is a repository for LLW and MLW from the operation of the nuclear power plants.
The repository -SFR- will be located at the Forsmark site 160 km north of Stockholm on the Swedish east coast. Construction started in 1983 and deposition of waste will start in 1988. The repository is built as rock caverns 50 m deep into the rock below the bottom of the sea about 1 km off the coast. The water depth at the site is about 5 m.

The total capacity is planned for 90000 m^3 of waste. SFR will be built in two phases - the first is under construction now and the second is planned for the end of the 1990 s.

As the SFR is constructed in hard granitic rock it will give valuable experience for any future repository built in similar rock. A detailed hydrogeologic investigation programme is carried through the whole excavation and construction phase. This programme will give valuable field data for assessing mathematical models for ground water movements.

FINAL REPOSITORY FOR HIGH LEVEL WASTE

The feasibility of final disposal of spent nuclear fuel and high levelwaste have been demonstrated by the work reported in the KBS-1 and KBS-3 reports, which have been accepted by the Swedish government. These reports showed how long-lived wastes <u>can</u> be disposed of by present-day technology and within the geological conditions existing in Sweden.

Considerable work remains however to be done to show, how these measures are to be realized in detail and in an optimum way.

The future work for realization and optimization of a safe system for final disposal of nuclear waste will comprise.

- Continued research and development work in order to further deepen the scientific knowledge, that constitutes the base for the performance and safety assessment.

- Studies and evaluation of alternatives to the methods and concepts investigated so far.

- Optimization of systems as regards technology, economy and use of resources against the improved scientific knowledge.

- Investigations for site-selection.

A basis for the planning of the R&D-work is the overall timetable for realization of a final repository for spent fuel, see Figure 2.

According to the "Act on Nuclear Activities" the R&D-programme shall be updated every three years and sent to the authorities for review. The first programme subject to such review must be submitted in September 1986.

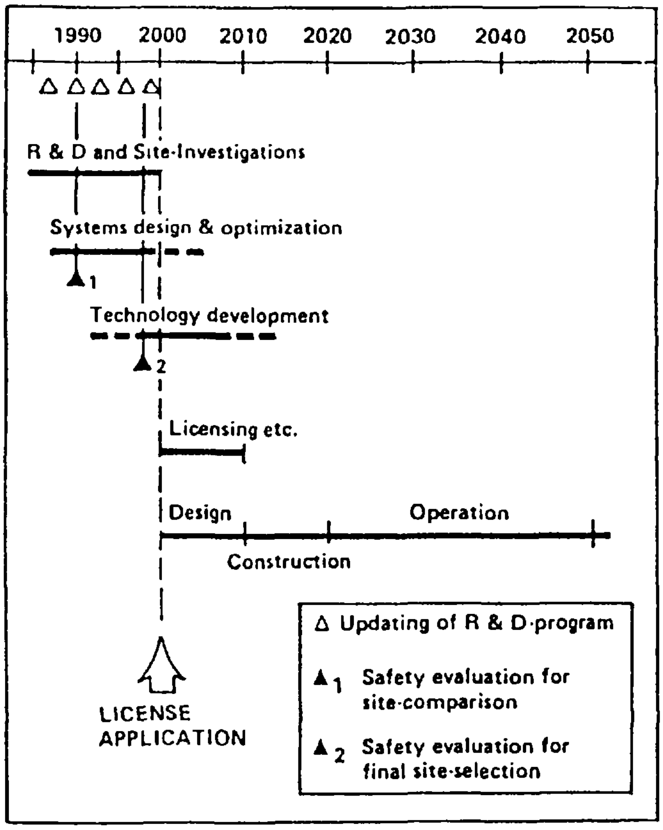

Figure 2. Overall time-schedule for realization of HLW-repository

RESEARCH AND DEVELOPMENT FOR SPENT FUEL AND HLW REPOSITORY

An important part of the ongoing research programme for a high level waste repository in Sweden are the site selection studies. These are described more in detail in another paper to this meeting by Hans Carlsson /1/.

Site investigations started in Sweden in 1977. The investigations so far follow a standard programme which is continuously updated as instruments and methods are further improved and developed. Eight sites have been fully investigated up to now according to this programme. An additional three to four sites will be studied up to 1990. Two or three sites will then be selected for more detailed studies during the 1990s. The final selection of a site for the repository will be made at the end of the 1990s.

Beside the site selection studies the geoscientific R&D is directed towards three major areas

- comprehensive studies of fracture zones
- studies of bedrock stability
- studies and modelling of groundwater movement and nuclidemigration

All these areas are of great importance for the final site selection and for the safety assessment.

The studies on chemistry are falling into four categories - ground water chemistry, radionuclide chemistry, release and transport model development and in-situ tests and natural analogue studies. The very long time frame that has to be considered in the performance assessment of a nuclear waste repository makes it attractive to supplement laboratory and field experiments with observations in nature, which reflect similar processes over geological time spans. An international Workshop on natural analogues was organized by SKB and the US-DOE in Chicago in October 1984 /3/. The main conclusion from the workshop was that the natural processes to be studied must be carefully selected and evaluated in close cooperation between the investigating geoscientist and the modeller of the corresponding processes in waste repository systems. The emphasis is on processes with well defined boundary conditions rather than complete geological repository systems. SKB is supporting studies on uranium mobility in granitic rocks, on Morro do Ferro and on evaluation of data from Oklo.

An important part of any R&D-programme for HLW disposal is the study of the waste form. The work in Sweden on spent fuel has been continuing since several years. Total contact times between spent fuel specimens and groundwater up to more than 900 days have now been achieved. Special studies of solubility constraints and the effects of radiolysis have been started. The measurements on uranium and plutonium indicate that a solubility limit is rapidly achieved and that this limit stays constant irrespective of contact time. Comparison between measured equilibrium concentrations and calculated solubilities show reasonably good agreement /2/

The studies of canister materials and buffer and backfill materials are continued along the lines followed in the studies for the KBS-reports. Future work will include investigations on alternatives to copper and bentonite, which were the main materials studied for KBS-3.

In the area of performance assessment and safety analysis the main emphasize in the recent years have been put on the development of a computer-code-system PROPER which is based on the main ideas which were first implemented in the AECL-code SYVAC.The new code is mainly intended for intercomparison studies concerning R&D priorities, design optimization, site screening and selection and safety assessments. In detailed safety analyses the code must be backed up by more elaborate and comprehensive methods.

INTERNATIONAL COOPERATION

The Swedish waste management program includes an extensive international cooperative framework. Sweden is the host country and SKB the executive organisation for the international OECD/NEA - Stripa-project. Nine countries are cooperating in this project, which uses an abandoned iron-ore mine at Stripa some 250 km west of Stockholm. The project is now in phase II which will be completed in 1986. The phase I and phase II studies include the following items

- hydrogeological and hydrogeochemical characterization of the site and the groundwaters including application of new techniques and new sampling methods,
- a major buffer mass test with electrically simulated heaters in half-scale deposition bore-holes,
- development and demonstration of cross-hole techniques for detection and characterization of fracture zones,
- 2D- and 3D-migration experiments,
- borehole, shaft and tunnel sealing tests.

Results from the phase I and the ongoing phase II were presented and discussed at a NEA-symposium in Stockholm in June this year. The planning of a tentative phase III starting in 1986 is now in progress.

SKB also participates in the joint Japan - Switzerland - Sweden (JSS) project for studies of HLW glass. Radioactive glasses were obtained from Cogema in France and are investigated at Studsvik in Sweden and at Würenlingen in Switzerland. Supplementary investigations on inactive simulated HLW glass are made at Stripa and in various laboratories.

SKB has bilateral information exchange agreements with US-DOE, CEA in France, AECL in Canada and NAGRA in Switzerland. An agreement with CEC-Euratom has been negotiated but not yet formally approved.

COSTS AND FINANCING OF RADIOACTIVE WASTE MANAGEMENT IN SWEDEN

All costs for radioactive waste management and decommissioning of nuclear power plants in Sweden has to be carried by the owners of these plants. The costs are covered by a fee on the nuclear power production. The fee is set by the government and revised each year. For 1984 and 1985 the fee is 0.019 SEK/kWh (about 2.2 US mills/kWh). The fee is based on cost calculations made by SKB and submitted annually to NAK. This authority reviews the calculations and proposes a fee to the government. According to the law the fees should be set individually for each reactor based on its individual projected costs. Sofar there has however not been a firm enough base for differentiating the fees but the same fee applies to all units. In the future it is likely that the fees will be differentiated between the different utilities. The fees are collected in a special fund at the Bank of Sweden. The fund is administered by NAK, which also submits money from the fund to the various waste management activities performed by SKB. Excluded from this financing is all facilities needed for the handling and disposal of reactor operational wastes. These facilities are financed directly by the utilities through SKB. In particular this is the case for the major part of SFR.

From the 1985 SKB-report to NAK on cost calculations the following figures were obtained /4/. The already spent costs were 3.4 GSEK covering CLAB, SFR, reprocessing services, R&D-program and the transport system. The calculated future costs (at price level January 1985) are 43.4 GSEK.

It should be kept in mind that many of the costs will occur in the far future. Figure 3 gives a rough account of the distribution in time of the future costs.

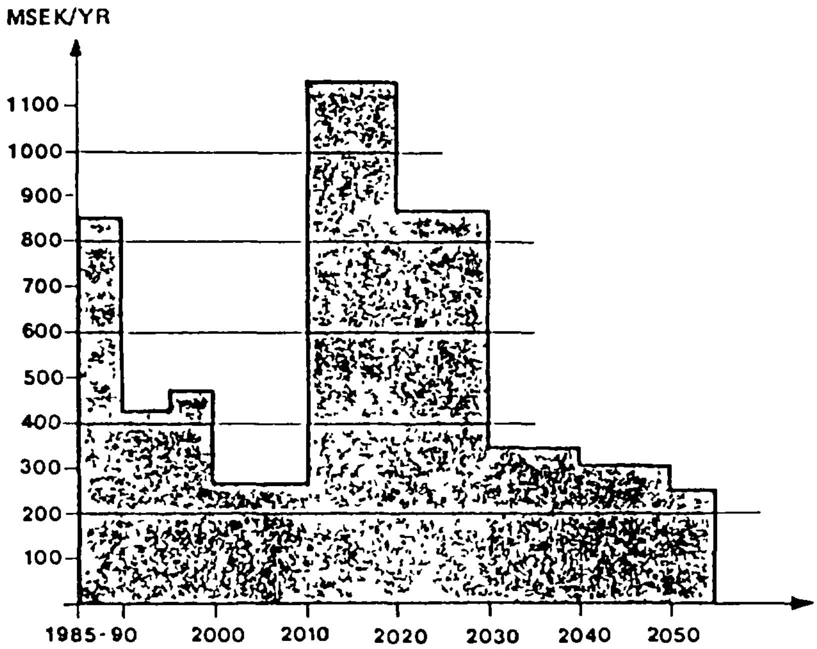

Figure 3. Approximate distribution of future waste management costs for Swedens nuclear power program

The relative distribution of the total costs not accounting for any interest or present value calculation are roughly according to the following table.

Interim storage of spent fuel and other wastes	18%
Reprocessing of 690 MTU	10%
Final disposal of spent fuel and long-lived wastes	31%
Final disposal of operational and decommissioning wastes	4%
Transportation of wastes	6%
Decommissioning and dismantling of nuclear power plants	24%
Miscellaneous including R&D	7%

The cost for direct disposal of spent nuclear fuel without reprocessing is calculated to about 3100 SEK/kg uranium including associated R&D-costs. Of this cost 7% is attributed to transportation, 30% to intermediate storage in CLAB for 40 years, 37% to encapsulation in copper canisters and 26% to final disposal according to the KBS-3-concept. A substantial part of the costs for spent fuel disposal is fixed costs. The marginal cost for additional quantities was calculated to 1300 SEK/kg U.

CONCLUDING REMARKS

The current Swedish programme for waste management is at present emphasized on

- start of operation of the interim fuel storage facility CLAB
- construction and completion by 1988 of the first phase of the repository for reactor operational wastes SFR
- basic research and development on disposal of high level waste with direct disposal of spent fuel as the prefered route
- continued site investigations for finding a suitable site for a HLW repository. A major screening and selection of 2-3 sites for more detailed studies will be made around 1990
- studies of alternatives to the methods described in the KBS-reports
- extensive international cooperation in several areas in particular through the Stripa- and JSS-projects.

The total costs for waste management (including decommissioning and dismantling of the nuclear power plants) are about 0.020 SEK per kWh which is about 10% of the consumer prize for electricity. Under present conditions in Sweden direct disposal of spent fuel is more economical than reprocessing.

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SPENT FUEL MANAGEMENT IN SWITZERLAND

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Abstract

Switzerland currently has 3 000 MWe being delivered from five nuclear power plants. Two more 1000 MWe power plants are firmly planned. The current spent fuel management and disposal programme including contracts for reprocessing of all spent fuel generated up to 1990 is presented. The plan for intermediate storage of spent nuclear fuel away from the nuclear power plants, as well as the storage of vitrified high-level wastes is given.

1. The Nuclear Program and its Policy for Radioactive Waste Management

Nuclear power was introduced already 16 years ago in Switzerland and it accounts today five nuclear power stations, with a total capacity of about 3000 MWe and two more 1000 MWe nuclear power plants firmly planned; in 1985 about 40% of the electricity produced in the country is of nuclear origin.
For final disposal, planning of a 6000 MWe nuclear energy programme was taken as a basis.Assuming 40 years as the operating period for each nuclear power plant,and assuming also that reprocessing of all the spent fuel elements could be done,the total amount of the arising waste works out at 1000 m3 of high-level,70000 m3 of medium/ low waste level, and 100'000 m3 of low level wastes.The legal regulations require the permanent and safe disposal of all these wastes in final repositories.

In 1979 the Swiss Nuclear Energy Legislation was amended.Since then one of the conditions for licensing nuclear power stations is that of a firm programme of action must be established and implemented which will provide a "Guarantee" that the management of all the resulting radioactive wastes and their final disposal can be accomplished. For nuclear reactors which are licensed already,the Federal Goverment has stipulated that such a "Guarantee" must be provided by the the end of 1985.

These provisions also cover the high-level wastes separated during reprocessing operations.Under the more recent contracts,these could be sent back to Switzerland; the Federal Government had to enter into an agreement that no obstacle will be presented to the return of waste from fuel cycle operations contracted in France and the United Kingdom.

The appropiate investigations and research have been carried out by Nagra (National Cooperative for the Storage of Radioactive Waste). The 1985 Project Gewähr (Guarantee) is already completed and presented to the competent authorities in eigth volumes and individual research projects reported on in separate 150 reference reports.

The aim of Project Gewähr 1985, as determined by official requirements, can be defined in three points :

- Sufficiently detailed projects must cover the technical feasibility of a final repository in Switzerland
- By means of safety analyses it must be shown quantitatively that final disposal of all categories of radioactive waste is currently realisable based on present-day knowledge an available technology without exposing the population to unreasonable radiation risks
- The data put into the safety analyses must be corroborated by investigations in Switzerland or abroad or must, in some way, conform to the technical and scientific state of knowledge. The data include geological, hydrogeological and material-technical parameters and the behaviour of radionuclides in the geosphere and the biosphere as determined by their physical and chemical properties.

Demonstration of technical feasibility in the context of project Gewähr is provided by constructional engineering project studies and that of log-term safety by relevant safety analyses. No repository locations are specified for Project Gewähr 1985 because, site selection is part of Nagra's longer term programme for implementation of repository projects.

In the independent Nuclear Energy Inspectorate attached to the Energy Ministry, a separate branch has been set up in order to handle the authorization of repositories.

2. Current Management and Disposal Programme

Contracts are held for the reprocessing abroad of all spent fuel up until at least the year 1990, and longer for some reactors. These contracts also cover the related transport and storage and the vitrification of the resulting high-level waste. Following its initial cooling in the reactor pools, the spent fuel is transported abroad and only medium and low-level reactor waste management is actually undertaken at this time within Switzerland.

2.1 Waste from Reprocessing Contracts

The high-level waste (HLW) and the remaining medium/low wastes (MLW/LAW) should be disposed off in two different types of repositories, as follows:

- Type C repository for high-level and certain alpha-containing intermediate-level waste, and
- Type B repository for all remaining intermediate- and low-level waste.

Using acceptance specifications, the waste is divided between the two repository types. Allocation and control procedures ensure that no waste with an unacceptably high activity concentration will be disposed of in the repository for low- and intermediate-level waste.

In the context of Project Gewähr 1985 the Type C repository is planned to be situated around 1200 m deep in a stable granite formation away from large fault zones in the crystalline basement of north Switzerland. The high-level waste (HLW) is disposed of underground in horizontally mined tunnels while final disposal of intermediate-level waste (ILW) is in vertical subterranean silos.

To provide for the final geological disposal of high-level wastes Type C, in the crystalline basement under the northern part of Swiss lowlands, a deep drilling programme was started for geological exploration in 1982. Selection of the final site is foreseen around 1995; construction of the final repository for high-level waste should start after the year 2000 and the facility is planned to be ready in 2020. The packaging of unreprocessed spent fuel for introduction into the repository is also being studied.

In Project Gewähr 1985 the Type B repository is planned as a mined cavern system with horizontal access tunnels in an alpine formation of Valanginian marl. The overburden will be at least 750 m. All waste is emplaced in standard repository containers into which the smaller waste packages are transferred in the reception area and then encapsulated with liquid cement.

To establish a medium- and low-level waste Type B repository, three sites, one in an anhydrite, one in a crystalline and one in a marl formation have been selected for further exploration by means of gallery construction. For the final repository at one of these sites, construction should begin in 1991 so that the facility can operate from the year 1995.

The current estimates of total waste arising from the reprocessing contracts abroad are as follows:

Reprocessing wastes		
HLW (m^3)	Alpha (m^3)	LLW/ILW (m^3)
240	4500	30000

2.2 Intermediate Storage

The nuclear utilities have set up an engineering consortium, the Consortium d'Etudes de Lucens (CEL), for planning the intermediate storage of spent nuclear fuel away from the nuclear power plants, as well as the intermediate storage of vitrified high-level wastes and other reprocessing wastes returned to Switzerland.

In the case that no new reprocessing contracts are available, the spent fuel already at the different nuclear power stations can be stored in a central facility.

The water-pool storage capacity at the different reactors reaches for 5 to 14 years of operation; any excess of spent fuel would be stored away from reactor. A project evaluation has shown the advantages of dry-storage, expecially in casks, depending on the size of the storage facility.

CEL is applying since 1985 for a general permit for a combined "away-from-reactor" storage facility for spent fuel, vitrified high-level wase and medium- and low-level waste from reprocessing. This facility is planned to be operational by 1992.

The intermediate storage facility is been projected to store HLW and/or spent fuel in dry transport and storage casks in a surface building. Low and medium active waste can be stored in separated adjacent facility to the previous one.

The storage capacity for spent fuel and HLW reaches from 30 to 50 years and can be commissioned in various stages e.g. for the next 15 to 20 years in order to take the HLW of the first reprocessing contracts starting around 1992.

The total design capacity of the storage facility can take 184 transport casks with HLW or 1555 tu of spent fuel non being reprocessed and in this way the capacity of the facility reaches untill year 2005; when taking into account the reprocessing option, the capacity is extended until year 2020.

3. Finance

NAGRA and CEL are non-profit organizations, financed by their participating members in accordance with their service needs. The overall cost of intermediate storage and final disposal, including research and development, is calculated to represent about 2% of the production costs of nuclear electricity. To the end of 1985 NAGRA has invested about 270 million Swiss Francs in its research programmes.

4. International Co-operation and Research

As well as participating in the International Atomic Energy Agency (IAEA) and the OECD's Nuclear Energy Agency (NEA), Switzerland has agreements for the exchange of information in this field with France, Sweden, the European Economic Community and Austria. In addition Switzerland participates in the NEA's Stripa project in Sweden and in some joint work with the Federal Republic of Germany. In 1985 Switzerland started its own research at the newly Commissioned Grimsel-Laboratory in the Southern Alps in order to intensify different geological research programmes in the frame of its project Gewähr (Guarantee).

IRRADIATED FUEL MANAGEMENT IN THE UNITED KINGDOM

National contribution presented by D. Groom
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Abstract

The United Kingdom has a long-established nuclear generating program starting in 1956. In 1985, 20% of electricity available from the UK public supply came from nuclear power. The current management strategy for the UK irradiated fuel and reprocessing waste is to: 1) continue to reprocess spent Magnox and AGR fuel separating waste products and the recovery of uranium and plutonium; 2) store high-level liquid waste until the Windscale vitrification plant comes on line later in 1980's; 3) store vitrified waste in engineered stores at the surface for at least 50 years before disposal; 4) develop scientific and technical knowledge on alternatives for disposal of HLW; and 5) proceed with establishing disposal routes for the LLW and MLW from reprocessing, nuclear power station operation and other industrial and medical uses.

All aspects connected with the national programme on storage, reprocessing, transport and future activities in this subject field are presented.

NATIONAL POLICY FRAMEWORK

1. The United Kingdom has a long-established nuclear generating programme, having been, in 1956, the first country to deliver electricity to the national grid from a commercial power station. It has for many years produced electricity commercially from both "Magnox Reactors" and "Advanced Gas-Cooled Reactors" (AGRs). In 1985, 20% of electricity available from the UK public supply came from nuclear power stations.

2. The UK Government's policy is to encourage the electricity supply industry to ensure that there are reliable, safe and cost effective reactor systems available for ordering as necessary. In 1977, the Central Electricity Generating Board (CEGB) declared its intention of establishing the PWR as a valid option for the UK. The Government, in confirming its agreement to this intention, took the view that subject to the necessary consents and safety clearances, a PWR should be the next nuclear power station order. The CEGB have placed an application to construct the UK's first PWR at Sizewell in Suffolk and this has been the subject of detailed examination at a Public Inquiry. The Inquiry closed in March 1985 and the Inspector's report to the Secretary of State for Energy is awaited.

3. The UK, together with France, FRG, Italy and Belgium, is a member of a European collaboration the purpose of which is to develop the fast reactor and its fuel cycle as an economic source of electricity for the future.

The collaboration, which was established in January 1984 through an intergovernmental Memorandum of Understanding, will allow the UK to pool efforts with those of our partners with a view to maximising our respective technical achievements and expertise whilst minimising duplication. The Memorandum is intended as an umbrella under which the industry will be able to progressively set up its own general and specific implementing agreements covering three broad fronts. These are R&D on fast reactors; their design construction and operation; and the related fuel cycle. Specific agreements concluded to date include:

a) A general agreement between CEGB and EdF on reactor design, construction and operation.

b) Memorandum of Understanding between BNFL and UKAEA for the UK, and CEA and COGEMA for France, on fuel fabrication and reprocessing.

c) A Memorandum of Understanding on reactor design and R&D between the UKAEA and NNC on the UK side and parallel organisations in France, FRG, Italy and Belgium.

In this context, an outline planning application for the building of a European Demonstration Reprocessing Plant (EDRP) has been made jointly by the UKAEA and BNFL. A public local planning inquiry into the proposal opened at Thurso, on the north coast of Scotland, on Monday, 7 April 1986.

4. Within this framework, the current management strategy for the UK's irradiated fuel and reprocessing waste is to:

(1) continue to reprocess irradiated Magnox fuel, and to reprocess irradiated AGR fuel in THORP when it becomes operational, allowing the separation of waste products and the recovery of uranium and plutonium.

(2) continue to store the high-level liquid waste from reprocessing at Sellafield in cooled stainless steel tanks pending operation of the Windscale Vitrification Plant in the late 1980's.

(3) store the vitrified waste in engineered stores at the surface for at least 50 years before disposal, to allow heat and radiation to reduce and to allow time for evaluation of options for disposal;

(4) continue to develop scientific and technical knowledge on options for final disposal of vitrified high level waste, for use when the time for a decision arrives; and meanwhile,

(5) to proceed with establishing disposal routes for the intermediate and low-level solid wastes resulting from reprocessing, nuclear power station

operation and other industrial and medical uses of radioactive materials;

(6) to continue to dispose of very small quantities of radioactivity in the form of low level liquid, atmospheric and solid wastes under the terms of authorisations granted by the Regulatory Departments DoE and MAFF.

INSTITUTIONAL RESPONSIBILITIES

5. National responsibilities for energy, radioactive waste management and transport policies are vested in the Secretaries of State for Energy, Environment and Transport respectively and the Secretaries of State for Scotland and Wales each of whom consults with other Ministries concerned with these subjects. The regulatory function is performed as appropriate through Inspectors of the Radiochemical Inspectorate of the Department of the Environment (DoE) and of the Ministry of Agriculture, Fisheries and Food (MAFF), Scottish Development Department and Welsh Office, and Officers of the Nuclear Installations Inspectorate (NII) of the Health and Safety Executive. The Government itself takes advice from independent expert committees, the Advisory Committee on the Safety of Nuclear Installations (ACSNI), the Radioactive Waste Management Advisory Committee (RWMAC), and the Advisory Committee on Transport of Radioactive Materials (ACTRAM), in formulating national policy. These committees include numbers of people drawn from Universities and other public bodies, including the Trade Unions.

6. Methods of transport conform to IAEA guidelines and are regulated, as stated above, by Department of Transport.

7. Operational responsibility lies with the nuclear industry. Two public electricity boards (The Central Electricity Generating Board, CEGB and the South of Scotland Electricity Board, SSEB), the United Kingdom Atomic Energy Authority (UKAEA), and the national nuclear fuel services supplier, British Nuclear Fuels plc (BNFL), own and operate reactors supplying electricity to the national electricity grid in the UK. These organisations are responsible for the safe management of the fuel used in their reactors, including discharge of irradiated fuel from the reactors, storage at the reactor site and transport of fuel for reprocessing.

8. BNFL is responsible for the design, construction and operation of facilities for the supply of nuclear fuel cycle services, including reprocessing of irradiated fuel and management of the resulting wastes. The Department of the Environment is responsible for research on methods of disposal of radioactive waste, including that from reprocessing, which it conducts through a variety of bodies, including the UKAEA, the Institute Oceanographic Sciences, the British Geological Survey and certain industrial firms.

9. In 1982 BNFL, the CEGB, the SSEB and the UKAEA set up an organisation, the Nuclear Industry Radioactive Waste Executive (NIREX) which is charged with securing the disposal of low and intermediate level wastes from all the partner organisations and, with appropriate charges, from other UK organisations. United Kingdom Nirex Limited was formed in November 1985 from the Nuclear Industry Radioactive Waste Executive which has in effect become the new company's engineering and technical branch. Partners in NIREX are now shareholders in UK Nirex Ltd. The Central Electricity Generating Board and British Nuclear Fuels plc each hold 42.5 per cent of the shares and the United Kingdom Energy Authority and the South of Scotland Electricity Board each own 7.5 per cent. The Secretary of State for Energy holds a Special Share on behalf of the Government and has the right to appoint two Directors.

FINANCE

10. The policy of the public electricity boards is to charge their consumers at rates that properly reflect the past and identified future costs of providing the electricity. Accordingly, provision is made over the operating lifetimes of nuclear power stations for the expected costs of reprocessing irradiated fuel, decommissioning the stations and treating, storing and disposing of the radioactive waste products that will arise.

11. The cost of BNFL's facilities and their operation are recovered in prices charged to BNFL's customers. Development and operating costs incurred by NIREX are to be borne by the shareholders and other users of NIREX facilities.

12. Research and development for the disposal of vitrified high level wastes, is covered by government funding, administered by the Department of the Environment.

IRRADIATED FUEL MANAGEMENT AND RADIOACTIVE WASTE DISPOSAL

Initial Storage

13. Irradiated fuel discharged from reactors is stored for a period at the power stations, to allow short-lived radioactivity to decay and heat output to fall, before it is transferred to the BNFL reprocessing site at Sellafield. The fuel is cooled in ponds at all the stations except one, Wylfa, where it is held in an air-cooled vault. At Sellafield the fuel is stored in ponds to allow further decay of radioactivity until it is reprocessed and to provide an operational buffer.

Transportation

14. For transport from the power stations to Sellafield, the fuel is contained in very heavy steel casks, which have been designed to meet the stringent standards recommended by the IAEA and adopted and enforced by the UK regulatory authority, the Department of Transport.

Irradiated Fuel Reprocessing

15. Fuel from the Magnox reactors has been reprocessed at Sellafield in Cumbria since the start of the nuclear power programme. A significant quantity of the uranium recovered from reprocessing Magnox fuel has been recycled into AGR fuel. The plutonium is being stored for eventual use in fast reactors, although the possibility of recycling it in thermal oxide reactors has not been dismissed. Irradiated core fuel from the Prototype Fast Reactor (PFR) at Dounreay is reprocessed by the UKAEA in a plant owned and operated by them at the Dounreay site.

16. The construction of a plant to reprocess oxide fuel from AGRs and LWRs, THORP was approved by Parliament following the Windscale Public Inquiry in 1977. It will be located at Sellafield and is scheduled to come into operation in 1990.

17. High-level liquid waste concentrate arising from reprocessing operations at Sellafield is currently stored by BNFL on the Sellafield site in high integrity stainless steel tanks. Each tank is fitted with cooling, purging, monitoring and safety systems. BNFL is constructing a facility for the vitrification of these high-level wastes and those which will arise later from oxide fuel reprocessing in THORP. This facility will incorporate the appropriate features of the French vitrification process which has been successfully demonstrated in the AVM plant at Marcoule. The Windscale Vitrification Plant is due to come into operation in the late 1980s.

18. Various conditioning and encapsulation techniques for application to intermediate level wastes are under detailed investigation by BNFL, the CEGB and the UKAEA. The preferred technique for BNFL wastes, which constitute the majority, is encapsulation in cement.

19. The Radioactive Waste Management Advisory Committee (RWMAC) have advised the Government that the fast reactor system would not call for a significant change from the waste management strategy adopted for the thermal reactor system.

Intermediate Storage

20. Containers holding vitrified waste from the reprocessing of UK spent fuel will be placed in an air-cooled store. In their second Annual Report, issued in May 1981, the RWMAC suggested that containment in an engineered storage system for at least 50 years may be the best way to deal with solidified high-level waste prior to disposal. This would permit radioactive decay to reduce the thermal and radiation emission from the waste which would greatly simplify the disposal operation. A subsequent Government White Paper, Command 8607 established this approach as a part of the UK's waste management policy.

21. The longer term storage of irradiated fuel is also under consideration. The development of such longer term storage options would provide flexibility in the timing of the reprocessing of irradiated oxide fuel beyond the THORP baseload. In this respect the CEGB is developing a design of dry buffer store suitable for the longer term storage of irradiated AGR fuel. Any PWR stations built in the UK will be equipped with ponds which will be capable of holding around 18 years worth of fuel arisings.

RADIOACTIVE WASTE DISPOSAL

22. As noted above, containment of vitrified high-level waste in engineered storage is currently envisaged for at least 50 years before disposal; this will allow completion of a research and development programme designed to provide information on options for safe disposal.

23. In contrast to high level waste, there is no technical advantage in delaying disposal of either LLW or ILW, and no technical barrier to their disposal. In its latest Annual Report, NIREX observed that only two new facilities are required to accommodate the entire volume of existing low and intermediate level wastes and those expected to arise in the UK over the next 50 years. It is giving attention to plans for:

- a deep repository (at least 100 metres underground on land, or deep beneath the sea bed) for intermediate level wastes.

- a shallow land burial facility for low-level wastes from a variety of medical, industrial, research and other sources, as well as from the operation of nuclear power stations.

Both types of repository involve encapsulation in a suitable matrix, engineered containment (concrete, steel), and properties of natural geological barriers, for example a clay formation.

24. On 24 January 1985 the secretary of state announced that NIREX would be required to carry out geological investigations of at least three possible sites for each type of disposal facility. On 25 February 1986 NIREX announced the location of four sites that it proposes to investigate to determine suitability for the disposal of low-level wastes. It is anticipated that in 1988 a public inquiry will be held before the preferred site can be developed. The disposal facility is targetted to become operational in about 1993. In parallel with these activities, the Department of Environment is currently carrying out a review of the technical options for the disposal of ILW.

Table 1

An Indication of AGR Fuel Discharges and Storage Capacities in the UK up to 1995

	Cumulative Totals to end of year shown (tU)		
	1985	1990	1995
AGR Fuel Discharges	550	1400	2500
Fuel Storage Capacity:			
At Reactor	200	250	250
At Reprocessing Site	1600	3000	3000

SPENT FUEL MANAGEMENT IN THE UNITED STATES OF AMERICA

National contribution presented by K. Klein
United States Department of Energy,
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Abstract

Spent fuel inventories in the USA exceed 12,000 t HM, and are increasing at a rate in excess of 1300 t per year. In the absence of reprocessing, these inventories are being stored in existing reactor fuel pools. Where initial pool storage capacity was insufficient, and structural and seismic considerations permitted, high density storage racks have been installed to increase pool storage capacity. In addition, a limited amount of spent fuel transshipment has occurred to other reactors within some utility systems where storage space was available. For the next few years there are not reactors in the United States whose continued operation is threatened because spent fuel storage space is not available or otherwise obtainable, using currently viable storage technologies.

Future plans for spent fuel management, as established by US law, call for continued utility management of their spent fuel until a Federal repository or Monitored Retrievable Storage Facility (MRS) is established. When one of these Federal facilities becomes available, high-level waste or spent fuel can be transferred to it and become a Federal responsibility.

The spent fuel management national programme and progress in this field are presented.

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1. INSTITUTIONAL FRAMEWORK

Commercial nuclear power was introduced into the United States in 1960 when the first fully private plant was completed. Full commercial status was attained in the late 1960s when plants larger than 500 MWe were brought into operation. Nuclear power has provided about 13 % of electricity generation since 1978 and is expected to provide approximately 19 % by 1995.

Institutional problems related to regulation, cost escalation and public acceptance, together with an abrupt, sharp reduction in electrical load growth, have resulted in no new plant orders since 1978. Most nuclear plants and a substantial fraction of fossil plants ordered over the past decade have been cancelled. Nearly all plants under construction over the past ten years have had their startup dates substantially deferred.

The original plan for a closed loop fuel cycle, with reprocessing of spent fuel within a year of discharge, has not been implemented. Following successful operations of the first commercial reprocessing facility in the United States, changes in licensing requirements and problems with innovative technology effected further commercial development of reprocessing. Reprocessing was then deferred indefinitely as a matter of national policy in 1977 pending an evaluation of proliferation concerns. This policy was reversed in 1981, at which time the private sector was encouraged to develop this part of the fuel cycle. However, remaining uncertainties associated with licensing, demand and future policies have inhibited private sector reprocessing initiatives to date.

As part of the United States nuclear development policy, substantially increased attention has been paid over recent years to the interim management and final disposal of nuclear waste and spent fuel. Following several years of debate, basic U.S. spent fuel management policies were finally established in a national law, the Nuclear Waste Policy Act of 1982. Implementation of this law is being pursued aggressively in accordance with a draft plan, called a Mission Plan, which was prepared by the U.S. Department of Energy (DOE) and is now being extensively reviewed before being made final and submitted to the Congress.

In addition, DOE is working with the nuclear industry and the Nuclear Regulatory Commission (NRC) to consider major changes to the regulatory structure and address other institutional issues that could prevent commercial nuclear power from realizing its full potential. Although nuclear power's contribution to electricity production is second only to coal and will double over the next decade, it has not reached its full potential in terms of displacing scarce natural energy resources. DOE programmes for cooperation with the industry and the Commission in dealing with institutional issues and programmes for reactor safety are designed to improve utilization of the nuclear resources.

Finally, United States nuclear development policy has shifted away from the rapid demonstration of commercial fast breeder reactors in that construction of the Clinch River Breeder Reactor demonstration plant has been cancelled. However, a strong R&D programme on all aspects of breeder development, including breeder fuel reprocessing, is being maintained.

2. CURRENT PRACTICE

Spent fuel inventories currently exceed 10 000 t HM, and they are expected to grow by 1 300 t in 1984 and at increasing rates thereafter. In the absence of reprocessing, these inventories are being stored in existing reactor fuel pools. Where initial pool storage capacity was insufficient, and structural and seismic considerations permitted, high density storage racks have been installed to increase pool storage capacity. In addition, a limited amount of spent fuel transshipping has occurred to other reactors within some utility systems where storage space was available. For the next few years there are no reactors in the United States whose continued operation is threatened because spent fuel storage space is not available or otherwise obtainable, using currently viable storage technologies.

3. FUTURE PLANS

Future plans for spent fuel management, as established by U.S. law, call for continued utility management of their spent fuel until a Federal repository or Monitored Retrievable Storage Facility (MRS) is established. When one of these Federal facilities becomes available, high-level waste or spent fuel can begin to be transferred to it and become a Federal responsibility. Prior to transfer to the Federal government, utilities may have spent fuel reprocessed if this service is available and they choose to utilize it.

Cumulative inventories of spent fuel in storage at reactor sites are expected to increase to about 25 000 t by 1990, 40 000 t by 1995 and perhaps 50 000 t by 2000. Operation of a geologic repository and acceptance of high-level waste or spent fuel for permanent disposal are scheduled to begin in 1998. Several years thereafter, the anticipated acceptance rate should exceed the generation rate. In the meantime, based on current utility plans, spent fuel inventories at many reactors will exceed current onsite storage capacity requiring many utilities to initiate plans for expanding capacity.

Continued expansion of onsite storage through established procedures, such as reracking and transshipping, will alleviate some but not all of these near-term storage problems; additional ways of storing fuel will be needed. A variety of research, development and demonstration efforts are underway including dry cask storage and rod consolidation programmes to establish additional onsite storage options and to streamline future licensing proceedings. These efforts include cooperative demonstrations involving the Federal government and private utilities.

The new U.S. law provides for a contingency in the event these additional storage technologies cannot be established soon enough at any reactor site. Until 1990, the DOE may contract with utilities for the interim storage of not more than 1 900 t of spent fuel. A full cost recovery fee would be charged, and users must have no reasonable alternative available to be certified eligible for this service by NRC.

The law also established MRS as an option for providing safe and reliable long-term storage. The DOE is to complete and submit to the U.S. Congress a detailed study of the need and feasibility of an MRS system. A proposal will be presented that will include detailed designs and plans for deployment. These facilities may, in conjunction with the geological repository, be an integral part of the overall waste management system, or they may be constructed as a contingency to ensure that acceptance of high-level waste or spent fuel can begin on schedule in the event the repository programme encounters significant delay.

4. FACTORS INFLUENCING SPENT FUEL MANAGEMENT POLICY

Prior to Federal acceptance of high-level waste or spent fuel for terminal storage, the primary decisions on spent fuel management will be made by utilities. These decisions will govern all aspects including methods, location, and timing of onsite spent fuel storage facility enhancements. These utility decisions will be strongly influenced by the economics of various storage alternatives, by the efficacy of Federal R&D efforts on them and by the NRC and public attitudes towards them.

The availability of domestic reprocessing is another factor that would affect spent fuel management policy. Reprocessing availability would be favoured by a strong demand for uranium and plutonium, but there is no indication of such a demand over the next few years. Consequently, commercial reprocessing is not expected to alleviate any major spent fuel storage problems of utilities over the next decade.

5. RESPONSIBILITIES AND FINANCIAL PROVISIONS

The Nuclear Waste Policy Act of 1982 reinforces the longstanding premise that the owners and generators of waste are responsible for paying the costs of disposal as well as for interim storage.

The law specifies that Federal disposal of high-level waste or spent fuel will be provided to utilities who enter into contracts for this service by 30th June 1983, or by the startup date for new reactors. A payment of one mill ($ 0.001) per kilowatt hour of electricity generated by nuclear plants after 7th April 1983 is made to a waste fund to cover the cost of constructing and operating disposal or long-term storage facilities. Equivalent payments are specified for spent fuel and for fuel in reactors on 7th April 1983. The fee is periodically reviewed and adjusted as necessary to ensure that revenues will cover the cost of the system.

Generic R&D on spent fuel storage is being continued using Federal funds. This programme has been greatly accelerated and expanded by the cooperative demonstration provisions of the Act. Under this programme utilities propose dry storage and rod consolidation development programmes that may be jointly funded with a 25 % limit on the Federal contribution.

Finally the Act provides for a Federal Interim Storage programme for storage of limited amounts of spent fuel that cannot be reasonably stored by the owner and present a threat to continued operation. The full cost of this service must be paid by the users and contracting authority extends only to 1990. The DOE is responsible for deploying the facilities to provide the spent fuel storage when needed and contracted. The NRC is responsible for determining that utilities who apply for this service are eligible under Commission rules before contracting may occur.

6. RESEARCH AND DEVELOPMENT ON SPENT FUEL MANAGEMENT

Generic spent fuel R&D is underway in many areas, including system performance, spent fuel integrity, analytical code development and dry storage in different atmospheres. A cooperative agreement is in force to test dry cask storage at a reactor site under NRC license at conservative conditions and, also, unlicensed at a Federal site where tests will probe the technical limits for dry cask storage. A second cooperative agreement is underway to test onsite dry storage in concrete silos. A third agreement to demonstrate fuel rod consolidation is under negotiation.

Research on the transportation of spent fuel is on-going, including planned testing of transportable storage casks. Research on the packaging of spent fuel for disposal as it relates to the various candidate media for geologic disposal is beginning with funding not separately identified.

Spent fuel information and the analysis of storage needs results in an annual report of expected storage space problem areas. This work supports periodic reviews of R&D planning and of the prospects for Federal Interim Storage needs.

An offer to participate in joint research and to provide technical assistance on spent fuel storage to non nuclear weapons states has been tendered as specified by the Act. Seven nations have responded to the tender offer.

Total Federal expenditures in 1984 for spent fuel R&D (excludes Monitored Retrievable Storage and Repository work) will exceed $ 6 million. Peak programme funding is expected to occur in 1985 at over $ 13 million. Funding is expected to decline thereafter and be phased out in the late 1980s as R&D work is completed and the technology is transferred. Private sector contributions to these programmes during this period are valued at about $ 50 million. Utility payments into the Waste Fund for repository, MRS and transportation activities are currently in excess of $ 300 million per year.

RECENT ACTIVITIES ON SPENT FUEL MANAGEMENT

OECD/NEA contribution presented by J. Vira
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Paris, France

Abstract

Activity of OECD/NEA in the field of spent fuel management is outlined. Recent studies on economics of the fuel cycle, status report on spent fuel management and study on decommissioning, feasibility, needs and costs are presented. Present activities cover a study on plutonium utilization in different reactors. Contents and goals of the publications prepared by NEA such as reports on projected nuclear fuel cycle requirements (Yellow Book), annual summaries of nuclear power and fuel cycle data (Brown Book) etc. are presented.

1. GENERAL

Work on the back-end of the fuel cycle is done in the Agency under the direction of several committees. Resource and economic aspects as well as general planning and systems considerations of the fuel cycle belong to the domain of the Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle (FCC). Technical aspects of waste management and disposal are studied under the Radioactive Waste Management Committee. An example of this latter work is the international Stripa project. However, in this presentation consideration is restricted to the work under the FCC.

2. RECENT STUDIES

In 1983 the FCC initiated three studies related to the fuel cycle back-end. The first of these was the experts group study on the economics of the fuel cycle. The work was finished during 1984 and the report was published in early 1985. The report discusses the levelized cost methodology for calculating the fuel cycle costs of nuclear power plants and also shows results from practical calculations for pressurized water reactors (PWR), heavy water reactors (HWR), and advanced thermal reactors (ATR). Two fuel cycle scenarios were defined for the PWR: the once-through cycle and the reprocessing cycle. A reference unit cost plus a range of variation for sensitivity checks were selected for each fuel cycle stage. Considerable emphasis was given to the costs of the back-end stages.

For the reference unit costs, the study indicated a difference between the back-end costs of the once-through cycle and the reprocessing cycle, in favour of the once-through option. However, in terms of total fuel cycle costs the difference was only of the order of 10 per cent, and if total electricity generation costs were considered, the difference would be no more than a few per cent. In addition to the reference case, results from various sensitivity calculations are presented in the report.

The second study is a status report on spent fuel management. It is also an experts report based on working group effort. The main objective of this study is to assess the current state of development of each stage of spent fuel management, including the disposal of high-level waste. For background, the report presents logistic data on spent fuel arisings and demand for capacity of fuel cycle facilities.

A substantial portion of the report is devoted to country-specific annexes which describe the spent fuel management situation in OECD Member Countries. Some OECD countries have already made a decision to reprocess part or all of their spent fuel. Other countries will postpone the decision. Because extended storage is considered safe and relatively simple, these countries can retain the reprocessing option for several decades. In finally deciding which option to choose, the countries may well end up with different choices, depending on various economic, strategic and general energy policy considerations.

The study is at an advanced stage, and the report is expected to be published within a few months.

The third study initiated in 1983 is the study on decommissioning, feasibility, needs and costs. The report of this study will be published before the summer.

3. PRESENT ACTIVITIES

The work programme for 1986 includes a study on plutonium utilization. According to the terms of reference this study will:

- consider the characteristics of alternative MOX fuel cycles for the PWR (self-generated fuel cycle, mixed systems etc.);
- define illustrative fuel cycle scenarios for MOX use;
- consider the economics of the MOX fuel use.

This study will be started within a few months and is expected to produce a technical report on the topic.

Another current task related to spent fuel management is the preparations for the International Symposium on the Back-End of the Fuel Cycle in spring 1987. The symposium will be jointly organized by the IAEA and the NEA.

4. STANDING COMMITMENTS

The work described above has been largely based on ad hoc working group efforts. In addition, however, the NEA has a number of standing commitments with bearing on the spent fuel management also. The NEA publishes regularly a report on projected nuclear fuel cycle requirements, the so-called Yellow Book. Two Yellow Books have already been published and the third one will come out later this spring. It discusses the demand for uranium and fuel cycle services, first in the short-term up to 1995, secondly in the long-term up to 2025. The analysis includes projections for spent fuel arisings and storage capacity requirements, taking into account the amounts of spent fuel that may be reprocessed in the future. The report is a joint activity with the IAEA and includes information also from outside the OECD countries.

Another regular activity is the publication of annual summaries of nuclear power and fuel cycle data ("Brown Book"). This short report presents statistical data and projections on nuclear electricity generation and planned capacities and requirements for fuel cycle services in the short-term in OECD countries. The next Brown Book will be issued in early May.

PART III
MAIN RESULTS OF THE ADVISORY GROUP MEETING

SUMMARY OF THE PAPERS

J.P. COLTON
Chairman

Advisers representing ten countries were invited: Argentina, Belgium, Czechoslovakia, France, Federal Republic of Germany, Japan, Sweden, Switzerland, UK and USA. OECD/NEA also sent an observer (see List of Participants) who presented NEA activities relating to spent fuel management.

The following are extracted from the country papers presented during the Advisory Group Meeting:

ARGENTINA

The current Argentine nuclear power program consists of HWR reactors: two in operation (Atucha-I, 345 MWe and EMBALSE, 600 MWe), one 745 MWe under construction and another 700 MWe to be installed before the end of the century.

With this size of program during the next 15 years the current plans for spent fuel storage are adequate and no plans for final disposal of fuel is planned. However, due to the limited uranium reserves and the high cost of fuel element production for the two HWR pressure vessel type reactors, an active program for the utilization of Mixed Oxide (U-235, Pu-239) fuel is being developed. The program allows the development of technology for reprocessing and MOX fuel fabrication on a pilot plant scale.

A plant for the reprocessing of nuclear fuels was designed and constructed by the Argentine Atomic Energy Commission with a capacity of 5 metric tonnes of uranium fuel per year. The fuel will be of the ATUCHA type ad the extracted Pu will be recycled into ATUCHA type fuel. The MOX fuel will be fabricated in a demonstration scale within special rooms in the reprocessing plant.

BELGIUM

Belgium currently has 5.5-7 GWe installed capacity with another 1-1.4 GWe planned but not yet committed. The spent fuel is stored at the reactor site in storage pools before being sent to a reprocessing plant. The plan for the spent fuel is to reprocess the fuel and recycle fissile products recovered into thermal and fast breeder reactors. Contracts exist with COGEMA for the reprocessing of approximately 600 tonnes U of fuel unloaded from Belgian power plants up to 1989.

CZECHOSLOVAKIA

Czechoslovakia currently has six WWER-440 reactors operating and six more units at different stages of their construction. It is planned that all 12 units will be operational by mid-1990's. During the next period, the WWER-1000 units will be introduced. Within the framework of a general bilateral agreement with the USSR on the cooperation in nuclear power development, it is agreed that the Soviet Union will accept the return of the spent fuel for reprocessing.

The original cooling pools allowed for 3 year at-reactor storage but due to subsequent needs and agreements away-from-reactor storage capacity for an additional 10 year storage is being constructed.

FRANCE

France's program is best characterized as a closed fuel cycle including reprocessing and use of breeer reactors. It has 38,958 MWe installed capacity with another 21,010 MWe capacity under construction. The chosen spent fuel management scheme is that of reprocessing.

Due to the large national program and the large contracted obligations at the La Hague reprocessing plant a large storage expansion program is underway (8,000t by 1988). This storage capacity allows for better buffer storage as well as better handling flexibility. To achieve maximum flexibility in the operation of the installations, the storage ponds, unloading units and reprocessing plants will all be interconnected through the ponds.

France has the most experience in the transport of LWR fuel. Transport of fuel has been made between the La Hague plant and 50 power plants in France, Belgium, FRG, Japan, Netherlands, Sweden and Switzerland. A total of more than 3400 tonnes of oxide fuel has been delivered in more than 1300 consignments. The current annual COGEMA managed, directly or indirectly, cask shipments involves approximately 300 casks either by road or by rail or by sea.

An intensive program for the recycling of recovered recycle material is underway involving the recycle into thermal reactors. This program has reached commercial level of application.

FEDERAL REPUBLIC OF GERMANY

The spent fuel management strategy in the FRG is based on interim storage and subsequent reprocessing of spent fuel. The waste shall be disposed of in a geologic repository. Parallel, alternative back-end fuel cycle techniques such as the direct disposal of the spent fuel without reprocessing are being investigated and might at a later date be used for the final disposal of spent fuel, which is not suitable for reprocessing.

Based on the estimated accumulation of spent fuel by the end of this century, the FRG back-end requirements will be met by reprocessing either in FRG or through contracts with COGEMA and BNFL. However, due to optimization of size and capability of the required facilities temporary storage capacity at nuclear power plant pools and additional AFR storage facilities play an important role in spent fuel management.

The interim storage of spent fuel within the FRG is accomplished primarily at-reactor with this storage being increased through the use of compacted storage racks. Dry storage is now a licensed technology in the FRG. Several types of CASTOR casks are now available for both storage and transportation of spent fuel. A 1500 MTU AFR facility has been completed and is waiting for final licensing. A second 1500 MTU facility is currently under construction. A request for a third facility co-located with the reprocessing plant has been submitted.

The cask consists of the following main components:
the cask body, a large piece of cast nodular iron
a closure system consisting of a double lid system
a basket system to hold the fuel assemblies
lifting trunnions.

The FRG also is recycling plutonium in LWRs as a method for using all excess plutonium which is not commited for immediate use in breeder reactors. Current studies indicate that industrial scale recycling is feasible.

Conceptual engineering has been carried out for a conditioning and encapsulation plant for use in direct fuel disposal. Licensing procedure for this plant will be started this year. The intention of this process is to dispose of fuel not suitable for reprocessing.

JAPAN

Japan now operates 33 nuclear power reactors with a combined capacity of 24,686 MWe. The future nuclear power generation capacity is forecasted to be about 34 MWe in 1990 and 62 GWe in 2000.

The basic nuclear spent fuel management concept is based upon the promotion of fast breeder reactors. Plutonium obtained from spent fuel through reprocessing will be used in FBRs, in advanced thermal reactors and also in LWRs. The reprocessing of spent fuel of LWRs is the chosen spent fuel management scheme selected by Japan. Current contracts for reprocessing is with foreign firms but a 800 tonnes reprocessing plant is planned for operation in mid 1990s. Due to this plan the spent fuel is being held in at-reactor pools and no away-from-reactor storage is being constructed. It should be noted that since the timing and amount of reprocessing is not yet firm, the storage duration and the amount of the spent fuels to be stored at reactor sites or at reprocessing plant sites is not yet fixed.

Studies of dry and wet storage methods are being made by both government and private companies.

Spent fuels from LWRs are shipped to the Tokai reprocessing plant by sea. Sea transportation is the most efficient way for Japan as both stations and reprocessing plant are located on the sea coast. Experience with the shipment of about 2700 tonnes of spent fuel to the UK and France has been successful.

SWEDEN

The nuclear power program in Sweden consists of 12 nuclear units. Their combined capacity is 9500 MWe.

Direct disposal without reprocessing is the main strategy for management of spent nuclear fuel in Sweden. Under the present

Swedish circumstances they have decided that it is the most economic route. Before final disposal the fuel will be stored for about 40 years in CLAB, specially built underground facility which was put in operation in July, 1985. This will give flexibility in the choice of final disposal method. Wastes from reactor operation will be disposed of in SFR, another underground facility in hard rock with planned operation starting in 1988. The total cost for waste management in Sweden is estimated to 47 billion SED (5.5 billion US$). This includes costs related to all types of waste and also the decommissioning of all twelve nuclear power facilities.

After the interim storage period the spent fuel will be encapsulated in corrosion resistant canisters and shipped to a final repository.

All nuclear power plants in Sweden are located at the coast and the sea is used for cooling. The interim fuel storage and the final LLW and MLW storage are also on the coast. As a result of their locations, sea transport has been selected as the main method to ship spent fuel and radioactive waste. Specially designed casks and ships are used for this movement.

The feasibility of final disposal of spent nuclear fuel and high level waste has been evaluated and accepted by the Swedish Government. These studies show that long-lived wastes can be disposed of by present-day technology and within the geological conditions existing in Sweden. Further work must be done to show realization and optimization of a safe system for final disposal of nuclear waste. This additional work consists of research and development, alternative methods evalution, optimization studies for best technology and economy, and site selection investigations.

Sweden is also the host country for the international OECD/NEA Stripa project. Nine countries are cooperating in this project which uses an abandoned mine at Stripa some 250 km west of Stockholm. The project includes: 1) hydrogeological and hydrogeochemical characterization of the site; 2) simulated heat buffer mass tests in half-scale deposit bore-holes; 3) rock fracture studies; 4) isotope migration experiments; and 5) borehole, shaft and tunnel sealing tests.

SWITZERLAND

Switzerland currently has 3000 MWe being delivered from five nuclear power plants. Two more 1000 MWe power plants are firmly planned.

The current spent fuel management and disposal program includes contracts for the reprocessing of all spent fuel generated up to 1990. The contracts also cover the related transport and storage and vitrification of the resulting high-level waste. The wastes from the reprocessing contracts will be returned and disposed of in two different types of repositories: Type C repository for high-level and certain alpha-containing intermediate level waste; and a Type B repository for all remaining intermediate- and low-level waste. Both of these repositories will be mined in underground facilities.

The nuclear utilities have set up an engineering consortium, to plan for the intermediate storage of spent nuclear away from the power plants, as well as the storage of vitrified high-level wastes and other reprocessing wastes returned to Switzerland.

UNITED KINGDOM

The United Kingdom has a long-established nuclear generating program starting in 1956. In 1985, 20% of electricity available from the UK public supply came from nuclear power. The current management strategy for the UK irradiated fuel and reprocessing waste is to: 1) continue to reprocess spent Magnox and AGR fuel separating waste products and the recovery of uranium and plutonium; 2) store high-level liquid waste until the Windscale vitrification plant comes on line later in 1980's; 3) store vitrified waste in engineered stores at the surface for at least 50 years before disposal; 4) develop scientific and technical knowledge on alternatives for disposal of HLW; and 5) proceed with establishing disposal routes for the LLW and MLW from reprocessing, nuclear power station operation and other industrial and medical uses.

Irradiated fuel discharged from reactors is stored for a period at the power stations, to allow short-lived radioactivity to decay and heat output to fall, before it is transferred to the BNFL reprocessing site. The fuel is cooled in ponds at all stations except one, Wylfa, where it is held in an air-cooled vault. At the reprocessing plant the fuel is stored in water filled ponds to allow further decay of radioactivity until it is reprocessed and to provide an operational buffer. The UK has moved over 10,000 casks safely over an average distance of 500 km.

An intermediate storage facility for longer term storage of the fuel is also under consideration. The development of such longer term storage options would provide flexibility in the timing of the reprocessing of irradiated oxide fuel in plants that will succeed THORP. In this respect the CEGB is developing a design of dry store suitable for the longer term storage of irradiated AGR fuel. Any PWR stations built in the UK will be equipped with ponds capable of 18 years worth of fuel arisings.

U S A

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Continued expansion of onsite storage through established procedures, such as reracking and transshipping, will alleviate some but not all of these near-term storage problems. A variety of research, development and demonstration efforts are underway including dry cask storage and rod consolidation programs to establish additional onsite storage options and to improve future licensing proceedings. These efforts include cooperative demonstrations involving the Federal Government and private utilities.

Generic spent fuel research and development is underway in many areas, including system performance, spent fuel integrity, analytical code development and dry storage in different atmospheres. A cooperative agreement is in force to test dry cask storage at a reactor site under NRC license at conservative conditions and, also, unlicensed at a Federal site where tests will probe the technical limits for dry cask storage. A second cooperative agreement is under way to test onsite dry storage in concrete silos. A third agreement to demonstrate fuel rod consolidation is under ngotiation.

Research on the transportation of spent fuel is ongoing, including planned testing of transportable storage casks. Research on the packaging of spent fuel for disposal as it relates to the various candidate media for geologic disposal is beginning with funding not separately identified. Total 1984 funding for spent fuel R&D (excludes MRS and Repository work) exceeded 6 million US$ and the 1985 budget was over 13 million US$. An offer to participate in joint research and to provide technical assistance on spent fuel storage to non-nuclear weapon States has been tendered as specified in the US, 1982 Waste Act. Cooperation is now ongoing with a number of different countries.

Finally, even though the United States nuclear development policy has shifted away from the rapid demonstration of commercial fast breeder reactors in that the construction of the Clinch River Breeder Reactor demonstration plant has been cancelled. However, a strong R&D programme on all aspects of breeder development, including breeder fuel reprocessing, is being maintained.

OTHER COUNTRIES

Several countries were invited to the Advisory Group Meeting and were unable to attend. Other countries have active programs but due to the limitation of attendance to the Advisory Group (Brazil, Bulgaria, GDR, Mexico, Republic of Korea, Spain etc.) were not invited. Summaries of their spent fuel management activities are found in documents from other meetings. Part I gives a partial list of meetings on topics on spent fuel management (1980-1986) were the various national approaches are discussed.

ADVISORY GROUP REVIEW AND RECOMMENDATIONS

The Advisory Group observed that spent fuel management continues to be of highest priority in assuring the optimum commercial use of nuclear energy. Spent fuel storage remains of crucial importance. It was observed that the spent fuel management strategy adopted by each country depends in detail on individual country factors.

Experience continues to show that water reactor fuel can be safely stored for extended periods of time.

A review of several country programmes shows that both commercial and prototype reprocessing facilities continue to operate satisfactorily.

Plutonium and uranium recovered by reprocessing represent a potential fuel resource. Plutonium is normally used in fast reactors but several countries with plutonium in excess of fast reactor requirements are actively pursuing recycle in thermal reactors. The utilization of recovered uranium leads to reduced uranium ore demand. Recovered uranium has already been recycled in several countries into thermal reactors and several other countries area actively developing a re-use capability.

It was agreed by the Advisory Group that past and current activities of the Agency have proven very beneficial in assisting them in matters relating to spent fuel management.

The members felt that the establishment of the Advisory Group for the review of and suggestions relating to IAEA future programmes of work in spent fuel management was very beneficial and should be reconvened in two years. It was suggested that the Agency might consider making the Group a permanent body in order to be continuously available for advice and consultation. It was suggested that the documents for review be sent to the advisers, a minimum of four weeks prior to the next meeting and that, due to the time consumed by the delivery and discussions of country status papers, a four-day meeting might be considered.

Good cooperation between the Agency and OECD/NEA was observed. The Advisory Group recommended that the Agency continue to note both national and international activities in spent fuel management in order to minimize duplication of efforts.

Review of 1987-88 Programme (these refer to items in Annex 1)

The Advisory Group suggested that the effort of the Meeting including the status reports under task 2 be issued as "Spent Fuel Management: Current Status and Prospects of IAEA Programme - Results of Second Meeting of Advisory Group on Spent Fuel Management".

The effort described in task 2 was commended by the Advisory Group and it was suggested that the new data on dry storage and rod consolidation be included.

The Advisory Group noted the efforts under the CRP (BEFAST) and encouraged the Agency to continue with BEFAST II. Due to the long planning cycle for both the Agency and the countries involved and the importance of continued surveillance of spent fuel, it is suggested that plans for the continued investigation should be encouraged.

In organizing the effort described in task 7 the Advisory Group encouraged the Agency to note the experience gained in BEFAST CRP.

It was proposed that the title of the effort in task 8 be changed to "Methodology for evaluating the economics of Wet and Dry Spent Fuel Storage".

Since the activities described in task 9 refer to similar previous activities of a meeting to be held in 1986, it was suggested that the results of this meeting be considered in the final preparation for the 1988 activities.

The Advisory Group suggested that the intent of the task described in task 10 be clarified by changing the title to read "Decontamination of transport casks and of spent fuel storage facilities".

The Advisory Group recommended that the results of the 1986 Technical Committee Meeting and the proposed consultants' meeting in preparation for task 9 be evaluated in order to establish if there is a need for any further activities.

It was suggested that the title of the activity described in task 13 be changed to "Status of MOX-fuel utilization in thermal reactors". The suggestion was also made that since OECD/NEA has similar activities in this area continued coordination should be made.

Under the topic of "Remote Technology" it was suggested that the Agency should focus on state-of-the-art evaluations and include areas under development where appropriate.

The Advisory Group suggested that both tasks described under items 15 and 16 should be combined and handled under a broader scope topic of

"Arisings and recovery of non-fissile useful materials from spent nuclear fuel and/or reprocessing wastes". It was noted that input from scientists with geological background may be required.

Advisory Group Suggestions for Future Activities

The following items were suggested for consideration in future IAEA programmes of work:

A. Current SFM practices and future plans in various countries (trends and analysis).

B. Management of spent fuel from research prototype reactors.

C. Review of operating experience and improvement of technology in reprocessing plants.

D. Including organizations/financing mechanisms for national spent fuel management options.

E. A review of summary of options available for each stage in Spent Fuel Management would be beneficial. This could include summary of country activities in each step.

F. Review the influence of higher burn-up on projected spent fuel arisings and on potential recycle and/or storage options.

G. Evaluation of the effect of multiple recycle on Spent Fuel Management.

H. To continue to advise and assist in furthering public knowledge on Spent Fuel Management issues.

I. Status of rod consolidation experience.

J. Status of recycled uranium utilization in reactors.

ANNEX 1

Division of Nuclear Fuel Cycle

Actions planned for 1987-88

Abbreviations

AGM - Advisory Group Meeting
CM - Consultants Meeting
CRP - Coordinated Research Programme
CRM - Coordinated Research Meeting
P - Publication
SE - Seminar
SY - Symposium
SP - Submission for publication
TC - Technical Committee Meeting
TRC - Training Course

Area of activity 1.2.4 - Spent Fuel Management

Project 1.1 Evaluation of information on spent nuclear fuel arisings and capacity requirements Task	Action or source	Services needed	Year of completion
1. Glossary of terms related to Spent Fuel Storage (French + Sp.versions)	CM 86 SP 87 P 87	Translation	1987
2. Techn.Document - Spent Fuel Management: Current Status and Prospects	CM 86 AG 86 P 87 CM 88 AG 88		1987 1989
3. Techn.Report - Updated Guidebook on Spent Fuel Storage	CM 86 CM 87 SP 88 P 89		1989
4. Training Course on Spent Fuel Storage	TRC 88		1988
5. Technical Cooperation Support & Missions		Technical Cooperation	Continuing

Project 1.2 To enhance international cooperation in selecting spent fuel storage options and practices Task			
6. CRP on Behaviour of Spent Fuel Assemblies and Storage Equipment at long-term storage conditions (1987-1991)	CM 87 CRM 88	data processing	1991

7. Techn.Document - Spent Fuel Surveillance and Monitoring Methods	TCM 87 P 89		1989
8. Techn.Document - Economics of Wet and Dry Spent Fuel Storage	CM 86 CM 87 SP 88 P 89		1989
9. Techn.Report - Improvement of structural materials resistance to chemical degradation and irradiation	CM 87 TC 88 CM 89 P 89		1989
10. Techn.Report - Decontamination of Spent Fuel Storage and Reprocessing Facilities	CM 88 TC 89 SP 89 P 90		1990
11. CRP - Behaviour of structural materials under irradiation with emphasis on heterogenious processes (1987-1991)	CM 87 CRM 87 CM 88	data processing	1991
Project 1.3 Exchange of information on spent fuel treatment the recycling of fissile materials and the recovery and utilization of other valuable elements Task			
12. Proc.of Intern.Symposium on the Back-end of the Nuclear Fuel Cycle: Strategy and Options	CM 86 CM 87 SY 87 SP 87 P 88		1988
13. Techn.Document - Status of MOX-fuel utilization in LWRs	CM 88 CM 89 SP 90 P 90		1990
14. Techn.Document - Remote technology of fuel fabrication with use of fissile recycled materials	CM 88 CM 89 SP 90 P 90		1990
15. Techn.Report - Noble metals arisings and demand	CM 87 CM 88 SP 89 P 90		1990
16. Technical Report - Sr and Cs arisings and demand	CM 87 TC 88 SP 89 P 89		1989

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