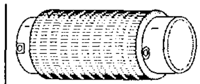
SPENT FUEL MANAGEMENT NEWSLETTER

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA



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FOREWORD

The first issue of the Spent Fuel Management Newsletter was published in March 1990. This publication was recommended by the IAEA Advisory Group on Spent Fuel Management as a part of the programme to keep Member States aware of the experience and developments in this field. The main purpose of the Newsletter is to provide Member States with new information about the state-of-the-art in one of the most important parts of the nuclear fuel cycle: the spent fuel management.

As in the first edition the contents of this publication consists of two parts:

- IAEA Secretariat activities - work and programme of the Nuclear Materials and Fuel Cycle Technology Section of the Division of Nuclear Fuel Cycle and Waste Management, recent and planned meetings and publications, Technical Cooperation projects, Coordinated Research programmes, etc.

- Country reports - national programmes on spent fuel management: current and planned storage and reprocessing capacities, spent fuel arisings, safety, transportation, storage and treatment of spent fuel.

The IAEA is grateful to all the authors who contributed to this Newsletter.

INTRODUCTION

Spent fuel management has always been one of the most important stages in the nuclear fuel cycle and it is still one of the most vital and common problems for all countries.

Continuous attention is being given by the IAEA to the collection, analysis and exchange of information on spent fuel management. Its role in this area is to provide a forum for exchanging information on the subject and to coordinate and to encourage closer cooperation among Member States in certain research and development activities that are of common interest.

In 1976 the International Atomic Energy Agency identified for the first time interim storage of spent fuel as an important independent step in the nuclear fuel cycle. The increased importance placed on spent fuel storage was caused by many factors; political, economic and technical difficulties in many IAEA Member States. The IAEA recognizes the importance of spent fuel management to many Member States and has been involved in related activities and studies.

The initial IAEA study pointing out the importance of spent fuel management was a working group called the "Regional Fuel Cycle Centre Study (RFCC)" in 1977. It was found in the RFCC study that while the front-end of the fuel cycle was well developed and was providing fuel for the various power reactors, the back-end of the fuel cycle was not yet fully developed in many countries. One of the most significant contributions of this study was the identification of the need for international cooperation and coordination of efforts. It also made it clear that very little information on the storage of spent fuel was available.

Since that time the IAEA has organized a number of projects in the field of spent fuel management. Some of the most important are:

- the Coordinated Research Programme on the Behaviour of Spent Fuel and Storage Facility Components During Long Term Storage (BEFAST)

- the Study of a World Survey of Spent Fuel Storage Experience for Wet and Dry Storage

- the Regular Advisory Group on Spent Fuel Management

- the Development of Safety Series Documents for Safe Long Term Storage of Spent Fuel

- Irradiated Fuel Management Advisory Missions (IFMAP)

The work related to the safety of spent fuel storage is listed as a high priority task in the IAEA Medium-Term Plan. As a new project in the Programme for 1993-1994, the Agency will offer to the developing countries a service to advise them on the long-term storage of spent fuel from nuclear power plants and/or research reactors in a technical cooperation programme called IFMAP - Irradiated Fuel Management Programme.

Coordinated meetings on the topic of spent fuel management were also organized in April 1982, with ANS/ENS in Brussels; in May 1982, with OECD and the Junta de Energia Nuclear of Spain in Madrid; in September 1983, with JEN in Madrid; in 1987 in Vienna a joint IAEA/NEA Symposium on the Back-End of the Nuclear Fuel Cycle: Strategies and Options; Conference of Nuclear Power Performance and Safety, in 1990 a joint IAEA/NEA Seminar on spent fuel storage: safety, engineering and environmental aspects and others.

Spent fuel management is the stage of the nuclear fuel cycle where the decision is made to follow an open approach, once-through cycle with permanent disposal of the spent fuel, or a closed cycle with reprocessing of the spent fuel and recycling of plutonium and uranium in thermal reactors or FBR's. Disposal involves steps which would place the spent fuel in location under conditions which would not allow for its removal. Reprocessing allows for the separation and recycle of the fissionable components (for reuse in reactors) from the waste material.

From the very beginning, the closed fuel cycle concept was to recycle the separated plutonium and uranium into breeder reactors. However, owing to delays and cancellation of breeder programmes in some countries, the separated fissile materials are being recycled into thermal reactors. At the present time thermal recycling is being implemented in Belgium, France, Germany, Japan, Switzerland and Russia. While other countries plan to have their spent fuel reprocessed by France, Russia or the United Kingdom, they have not committed themselves to the use of MOX fuels in their reactors.

A third approach is interim storage, with the possibility of monitoring the storage continuously and retrieving the spent fuel later for either final disposal or reprocessing. At the present time much attention is being given to this option in almost all the countries which have nuclear programmes.

In accordance with an IAEA study, in 1992, spent fuel arisings from all types of reactors amounted to over 10000 t HM, giving worldwide an estimated cumulative total of over 130000 t HM and projections indicate that the cumulative amount of spent fuel generated by the year 2000 may reach 230000 t HM. Assuming that only part of it is reprocessed, the amount to be stored is expected to be more than 140000 t HM. At the same time the first demonstrations of the direct disposal of spent fuel and HLW are expected only by the year 2020, therefore medium and long-term storage will be a primary option for the management of spent fuel during the next decades. In general, the design, technological, economic and material problems of safe spent fuel management will remain one of the important tasks of the nuclear community.

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PAST AND FUTURE IAEA SPENT FUEL MANAGEMENT ACTIVITIES

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The main objectives and strategies of the Agency's activities in the area of spent fuel management are to promote the exchange of information between Member States on technical, safety, environmental and economic aspects of spent fuel management technology, including storage, transport and treatment of spent fuel, and to provide assistance to Member States in the planning, implementation and operation of nuclear fuel cycle facilities.

During 1990, 1991 and 1992 a number of international meetings were organized and publications were issued (Appendix A) in accordance with the recommendations made by the Regular Advisory Group at its 1990 and 1991 meetings.

Below are listed the meetings which have been held since the last issue of the Spent Fuel Management Newsletter:

1. The International Seminar on Spent Fuel Storage - Safety, Engineering, Environmental and Economic Aspects

This Seminar was organized jointly by the International Atomic Energy Agency and the Nuclear Energy Agency of the OECD and held in Vienna in October 1990. The seminar was attended by more than 100 specialists from 30 countries and 4 international organizations and 36 papers were presented. It provided a forum for exchanging information on the state-of-the art and prospects of spent fuel storage with emphasis on safety, engineering, environmental and economic aspects as well as providing developing countries with the possibility of receiving information on experience and knowledge in this field.

2. Regular Advisory Group on Spent Fuel Management

A meeting of the Regular Advisory Group on Spent Fuel Management was held in Vienna in October 1991 to review the world-wide situation of spent fuel management, to define the most important directions of national efforts, and international cooperation in this area, to exchange information on the present status and progress in the performing of the nuclear fuel cycle, and to assist the IAEA to formulate the future programme in the subject field. 16 participants from 11 countries took part in this meeting. The Advisory Group observed that the activities related to the management of spent nuclear fuel continue to be of high priority in assuring the optimum safe use of nuclear energy. It was agreed by the participants that past and current activities of the IAEA have proven very beneficial in assisting countries in matters related to spent fuel management.

3. Technical Report on Strategies, Options and Trends in Spent Fuel Management

The Advisory Group Meeting (AGM) held in April 1991, with participants from 17 countries, discussed the plans of the Agency to prepare a Technical Report Series document on spent fuel management. In 1992 consultants from France, Germany, UK and the USA using the data from the AGM prepared the draft of the Technical Report Series (TRS) "Strategies, Options and Trends in Spent Fuel Management". The document presents the options and choices which are available for the management of spent nuclear fuel, and enables Member States to achieve an integrated approach to spent fuel management by addressing the important considerations in a structured manner. The TRS will be finalized in 1993.

4. Fuel Rod Consolidation

According to the recommendations made by the Regular Advisory Group on Spent Fuel Management in 1990, the Agency prepared a Technical Document (TECDOC) on "Consolidation of spent fuel rods from LWR". During two consultancies in November 1990 and June 1991 experts from US and Germany reviewed the state-of-the art of rod consolidation, compiled new information on the subject and defined the areas for future developments. The document presents the current status of the process by which Light Water Reactor (LWR) spent fuel is consolidated for the purpose of decreasing the space needed for interim storage and final disposal. The report outlines the major technical and licensing concerns, past and future demonstration of rod consolidation.

5. Establishment of Database on Spent Fuel Inventories, Projections and Characteristics

During the Advisory Group Meeting on "Spent Fuel Documentation, Inventories and Projections" in June 1992 participants from 10 countries elaborated a spent fuel management data base and evaluated an adequate set of information parameters to ensure safe and reliable handling of the spent fuel. The Group reviewed status reports from the countries represented, on spent fuel inventories, their projections and related documentation. The Group identified the main categories of data requirements relating to each of the back end stages of the fuel cycle, including handling and transport. An agreed set of data requirements and associated tracking arrangements which were judged to be the minimum necessary to support safe spent fuel management operations were established. The Group recommended that it should be proposed to the Member States, at a national level, to establish and maintain data records on spent fuel according to the format prepared by the IAEA. The preparation of the Technical Document on the subject will be continued in 1993.

6. Technical Report "Manual on Design, Technology and Operational Experience"

The main goal of this publication is to provide the specialists with new information and international experience on how to achieve technical requirements for spent fuel storage facilities, examples of methods, how to fulfil technical requirements will be discussed based on experience of different countries. Main principal components of the spent fuel storage facilities will be described separately with detailed explanation. The contents of the document and synopsis of some chapters were prepared during an Advisory Group meeting in 1992, attended by participants from 10 countries. Based on the outcome of this meeting, consultants from France, Russia, UK and the USA prepared the draft of the Technical Report. The TRS will be finalized in 1993.

7. Coordinated Research Programmes (CRPs) on Behaviour of Spent Fuel and Storage Facility Components During Long Term Storage (BEFAST-II, BEFAST-III)

The CRP on Behaviour of Spent Fuel and Storage Facility Components During Long Term Storage (BEFAST-II) has been completed. 12 countries with 16 agreements participated in this CRP. The third and final Research Coordination Meeting on BEFAST-II was held in March 1991 in Vienna. Participants from 11 countries took part in this meeting. During the meeting the last draft of the BEFAST-II Final Report was reviewed, the conclusions of the BEFAST-II Programme and eventual proposals for future IAEA activities were prepared. It was agreed by all the participants that it will be useful to initiate a BEFAST-III CRP continuing the activities.

In October 1992 14 participants having contracts with the IAEA, and 3 observers from a total of 12 countries participated in the first Research Coordination Meeting (RCM) of the BEFAST-III CRP, organized by both Atomic Energy of Canada and Ontario Hydro (OH) at OH's Headquarters in Toronto.

The purpose of the first RCM was:
- to report the status of national activities and the results achieved during the first year of the CRP,
- to discuss the subjects to be reported during the next Research Coordination Meetings,
- to finalize the table of contents of the future TECDOC, containing the BEFAST-III final report.

The Coordinated Research Programme is covering the period between 1991 - 95. The final report will be published in 1996.

8. Irradiated fuel management Advisory Missions (IFMAP)

A Consultants' Meeting was held to discuss the IAEA programme to initiate irradiated fuel management (IFMAP) services. The goal of the project is to assist the developing countries in questions related to the safe storage and management of fuel both from research and power reactors. A Pamphlet was prepared to inform the interested Member States about the possibility of having advice on irradiated fuel management in the framework of a Technical Co-operation (TC) project.

At the request of the Hungarian Atomic Energy Commission a team of German, Spanish, US, and IAEA experts went to Hungary to advise the Paks Nuclear Power Plant operators on the methods and available technologies of safe spent fuel storage. The experts provided up-to-date information to the operators and to the representatives of different Regulatory Agencies.

At the request of the Ukrainian State Committee on Nuclear and Radiation Safety, a Fact Finding Mission took place to discuss the problems associated with the management of radioactive wastes and spent fuel in the Ukraine. As a result of the Mission the IAEA has now a clearer view of Ukrainian needs, of the priorities and how to address these issues.

At the request of the Government of Thailand, through its Office of Atomic Energy for Peace, a mission took place to investigate conceptual designs of irradiated fuel storage facilities to meet the needs of the Thai research reactor programme, now and in the future. Several conceptual designs were recommended involving pool storage or dry storage.

9. Spent Fuel Storage Safety Series

According to the Programme and Budget of the IAEA for 1991-92, a Consultants' Meeting and subsequently an Advisory Group Meeting was held in 1990 to prepare and discuss the proposed structure of the Safety Series documents to cover all aspects of the storage of spent nuclear fuel.

It was agreed to prepare the following documents:

- Safety Guide on the Design of Spent Fuel Storage
- Safety Guide on the Operation of Spent Fuel Storage
- Safety Practices on the preparation of Safety Analysis Report for Spent Fuel Storage.

The documents are intended to give guidance on the key safety aspects of the long term, safe storage of spent nuclear fuel. They cover all relevant issues of the design, operation and licensing of interim spent fuel storage facilities. For each of the 3 documents a separate Working Group was formed. Each Group had meetings during 1991 and 1992, and produced a draft of their respective document, which was further discussed during a Technical Committee Meeting in Vienna in November 1992.

The TCM participants reviewed all the 3 prepared drafts. The nature of their comments was mainly editorial. The participants noted, that the contents of the documents are very well founded, and no omissions were found. The general opinion was, that the documents are in an advanced state, and that it does not seem necessary for their completion to have separate meetings for each working group.

The final draft of the documents will be reviewed by the IAEA Safety Series Review Committee, all comments will be incorporated in the text and will be sent out to the participants of a Technical Committee Meeting on Safe Long-term Storage of Spent Fuel. At present that TCM is scheduled for November 30 - December 3 1993, but at a later stage if the preparations are well advanced, the TCM probably can be brought forward.

10. Technical document "Catalogue of Methods, Tools and Techniques for Recovery from Fuel Damage Events"

On the basis of the recommendations of an Advisory Group Meeting (AGM on "Main Principles of Safe Management of Severely Damaged Nuclear Fuel and other Accident Generated Waste", 13-16 November 1989), the IAEA initiated a programme in 1990 to collect technical information on special tools and methods to deal with circumstances beyond the normal design basis of fuel damage.

A Questionnaire was sent out to solicit information from the countries and organizations which might have experience in this field. The responses to the Questionnaire were discussed at a consultants meeting and at an Advisory Group Meeting during 1990.

On the basis of this material the TECDOC-627 "Catalogue of Methods, Tools and Techniques for Recovery from Fuel Damage Events" was prepared.

The aim of publishing this document is to disseminate the experience gained in the Member States serving article 5 of the "Convention on Assistance in Case of a Nuclear Accident" and also to fill a potential void in response to fuel damage events of less severe magnitude.

11. Storage of Spent Fuel from Research and Test Reactors

Concerns in this area are mounting rapidly because of the refusal of some fuel vendors to take back spent fuel and the expense and added problems of waste disposal associated with commercial fuel reprocessing. Many countries without nuclear power programmes do not have the infrastructure to take back radioactive waste and are faced, at the moment, with only one viable option, i. e., to expand their existing interim storage facility or to build an extra facility. A programme has been started to address these special problems of the storage of irradiated fuels from research and test reactors. The first step, already underway, is to assess the full extent of the problem. Further steps, already planned, involve the initiation of a CRP and a training course on irradiated fuel storage from both power and research reactors.

12. Separation and Utilization of Caesium and Strontium from High Level Waste

A status report on the "Feasibility of Separation and Utilization of Caesium and Strontium from High Level Waste" was prepared together with experts from six countries, namely, Belgium, Germany, Japan, United Kingdom, USA and USSR.

The report is intended as a status report to provide a basis for further consideration of the options for Cs and Sr recovery and for the formulation of appropriate strategies. It examines:

- the present and future market demand for Cs and Sr;
- the technological feasibility of their separation;
- the possible impacts on high level waste management resulting from their separation;
- economic aspects;
- safety, environmental and public acceptability aspects of their utilization.

The document may be of interest to policy makers in the nuclear fuel cycle and in waste management.

13. Partitioning and Transmutation (P&T) of Actinides and Selected Fission Products from HLW

A meeting of the Advisory Group on "Partitioning and Transmutation of Actinides and Selected Fission Products from HLW" was held in October 1991 to review the status of research and development activities, to obtain information of the results of national and international studies on this topic, to define the most important problems in the field of partitioning and transmutation of actinides and fission products in relation to HLW management policy and to provide advice to the Secretariat on possible future IAEA activities in this field which may be better coordinated within the framework of international cooperation. 23 participants from 14 countries and 2 international organizations took part in the meeting. The Advisory Group observed a wide interest among participating countries in the partitioning and transmutation option as a possible complement to the reference scenarios of the back-end of fuel cycle comprising: storage of spent fuel, reprocessing and disposal of vitrified HLW into a deep geological repository. It was agreed that current and future activities of the IAEA could be very beneficial in assisting Member States in matters relating to the partitioning and transmutation programme and information exchange. The Group reviewed and agreed on the proposed 1993/94 programme.

In 1992 the 6 experts from Belgium, France, Japan, Russia and CEC reviewed the status of development in Member States in the field of partitioning and transmutation. It was pointed out that the existence of programmes being implemented by OECD/NEA and CEC led IAEA to establish a complementary programme on the fundamental safety aspects of partitioning and transmutation, that could be beneficial in assisting Member States in their national P&T programmes and fostering the information exchange.

Therefore, it was recommended that IAEA should undertake a study with emphasis on the environmental and non-proliferation implications of P&T. The scope of the CRP, proposed by the IAEA and the programme of the TCM in 1993 were discussed and agreed upon.

14. World Survey of Spent Fuel Treatment

The Advisory Group Meeting on "Spent Fuel Treatment and Emerging Problems in this Area" in October 1992 was attended by participants from 6 countries. The number of participating Member States was limited, and the AGM was held with the only objective to review the status and trends of spent fuel treatment, because of some concerns from the Member States about this sensitive subject, from the proliferation point of view. Country reports were presented and main areas of current development in spent fuel reprocessing were pointed out. The Advisory Group agreed that the IAEA should keep under review the state of the art in fuel reprocessing. Meetings for the exchange of information are useful, and should be organized at approximately 3-4 years' intervals. The proceedings of the meeting have been published as Working Materials.

15. Irradiation Degradation of Materials in Spent Fuel Storage Facilities

Degradation of the mechanical and physical properties of ageing materials in ageing irradiated fuel storage facilities, is beginning to raise serious concern. In particular, the lack of understanding of the fundamental mechanisms of a material's response to the corrosive environments found in irradiated fuel storage facilities raises a formidable barrier to the prediction of behaviour over extended time periods. This inability to extrapolate materials' behaviour with any confidence may cause problems with license extension for some facilities. Applications for license extensions well beyond the original design life may be prompted by the decision in some countries to cancel the fuel reprocessing option and delays in the availability of final disposal facilities in almost all countries. To address these concerns the first steps have been taken to set up a CRP on the topic of irradiation degradation of materials in spent fuel storage facilities.

16. Cost Analysis Methodology of Spent Fuel Storage

The report on "Cost Analysis Methodology of Spent Fuel Storage" was prepared. The reason for this project was that comparisons too often improperly present the relative costs of different spent fuel storage options because an appropriate methodology has not been used. One common error is attempting to compare assessments of different spent fuel management strategies undertaken by different nations where the requirements and circumstances are quite different. This report has been written to inform professionals involved in the development and implementation of policy decisions as well as staff who may be technically aware of, but not experienced in, the details of spent fuel storage. Furthermore, this report should also be useful for experienced nuclear engineers.

Taking into account that spent fuel management will be within the next decade a key issue for the nuclear fuel cycle and an important and common problem for the nuclear community, the Agency plans to continue its efforts in this field. The following meetings are scheduled for 1993:

1.	TCM on Away-from-Reactor Storage Concepts and their Implementation	March
2.	TCM/Workshop on Management of Fuel Failure Events	September
3.	TCM on Partitioning and Transmutation of Actinides and Fission Products	November
4.	TCM on Safety Guides for Long Term Storage of Spent Fuel	November
5.	AGM on Storage Experience with Fuel from Research Reactors	May
6.	AGM on Spent Fuel Management: Current Status and Prospects	September

Furthermore, the Agency is considering holding jointly with OECD/NEA a Symposium on Spent Fuel Storage, Safety, Engineering and Environmental Aspects in October 1994.

TCM = Technical Committee Meeting

AGM = Advisory Group Meeting

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REGULAR ADVISORY GROUP ON SPENT FUEL MANAGEMENT

The Regular Advisory Group on Spent Fuel Management (RAGSFM) was established in accordance with the recommendations of the Expert Group on International Spent Fuel Management in 1982. It has held meetings in 1984, 1986, 1988, 1990 and 1991. The RAGSFM is a Regular Advisory Group organized within the framework of the IAEA.

The Advisory Group consists of nominated experts from countries with considerable experience and/or requirements in such aspects of the back-end of the fuel cycle as storage, safety, transportation and treatment of spent fuel. The country membership is selected in such a manner as to reflect the various spent fuel management options ranging from the "closed" fuel cycle to "once-through" concepts and includes representatives from both developed and developing nuclear power users. The OECD/NEA is also a invited member.

SCOPE

The RAGSFM activities cover the following main topics:

a) Analysis and summary of spent fuel arisings and storage capacities;

b) Interface between spent fuel storage and transportation activities,

c) Spent fuel storage process and technology and related safety issues:

d) Treatment of spent fuel.

OBJECTIVE OF ACTIVITY

To provide technical advice to the Secretariat regarding the Agency's programme in the back-end of the nuclear fuel cycle;

To serve as a means of exchanging information on the current status and progress of national programmes;

To discuss and review the Agency's publications in this field;

To assist in the coordination of international activities in the back-end of the nuclear fuel cycle.

METHODS OF WORK

The Advisory Group meets every two years in order to:

- review and comment on the present Agency's activities in the area of the back-end of the nuclear fuel cycle;

- discuss the participants' presentations on the national current situation and future plans;

- define the most important directions of national efforts and international cooperation in the area of the back-end of the nuclear fuel cycle;

- prepare recommendations for future IAEA meetings and other related activities in the back-end of the nuclear fuel cycle.

The group's work between meetings is carried out and coordinated by the Scientific Secretary in cooperation with the Chairman on the basis of the RAGSFM meetings' recommendations. The Scientific Secretary keeps the group members informed of any important changes.

The results of each RAGSFM meeting are published as a TECDOC "Spent Fuel Management: Current Status and Prospect". The TECDOC usually includes:

a) Summary of Agency's spent fuel management programme,
  - publications
  - meetings
  - other related activities in the Agency

b) Country status reports

c) Summary and recommendations.

APPENDIX A

IAEA PUBLICATIONS ON SPENT FUEL MANAGEMENT

IAEA-TECDOC-345/R	IAEA Spent Fuel Glossary - Russian Edition	1986
IAEA-TECDOC-408	The Nuclear Fuel Cycle Information System/ An International Directory of Nuclear Fuel Cycle Facilities	1987
IAEA-TECDOC-414	Behaviour of Spent Fuel Assemblies during Extended Storage	1987
IAEA-TECDOC-418	Long-Term Wet Spent Nuclear Fuel Storage	1987
IAEA-TECDOC-419	Spent Fuel Management: Current Status and Prospects of the IAEA Programme	1987
STI/PUB/738	Back-End of the Nuclear Fuel Cycle: Strategies and Options (Proc.Int.Symp.)	1987
IAEA-TECDOC-421	Materials Reliability in the Back-end of the Nuclear Fuel Cycle	1987
STI/PUB/761	Nuclear Power Performance and Safety (Proc.Int.Conf.)	1988
IAEA-TECDOC-345/S	Glosario del OIEA sobre Terminos de Almacenamiento de combustible gastado	1988
IAEA-TECDOC-461	Spent Fuel Surveillance and Monitoring Methods	1988
IAEA-TECDOC-345/F	Glossaire de l'AIEA sur le Stockage du Combustible Irradie	1988
STI/PUB/751	Nuclear Power and Fuel Cycle: Status and Trends 1988	1988
STI/DOC/10/290	Survey of Experience with Dry Storage of Spent Nuclear Fuel and Update of Wet Storage Experience	1988
IAEA-TECDOC-487	Spent Fuel Management: Current Status and Prospects	1988
STI/PUB/749	The Nuclear Fuel Cycle Information System. A Directory of Nuclear Fuel Cycle Facilities	1988
IAEA-TECDOC-513	Management of Spent Fuel from Research and Prototype Power Reactors and Residues from Post-Irradiation Examination of Fuel	1989
STI/DOC/10/308	Feasibility of the Separation and Utilization of Palladium, Rhodium and Ruthenium from High-Level Waste	1989
IAEA-TECDOC-556	Decontamination of Transport Casks and Spent Fuel Storage Facilities	1990
IAEA-TECDOC-559	Methods for Expanding Capacity of Spent Fuel Storage Facilities	1990
IAEA /SPM/NL/1	Spent Fuel Management Newsletter	1990
IAEA-TECDOC-580	Spent Fuel Management: Current Status and Prospects	1990

STI/PUB/864	Nuclear Power, Nuclear Fuel Cycle and Waste Management: Status and Trends IAEA Yearbook 1990 (part of the SFM)	1990
IAEA-Working Material ISRS-SR-171	Spent Fuel Storage - Safety, Engineering and Environmental Aspects (Proc. of the Int.Seminar Vienna, Oct. 1990)	1990
STI/DOC/10/321	Management of Severely Damaged Nuclear Fuel and Related Waste	1991
STI/DOC/10/240 2nd Edition	Guidebook on Spent Fuel Storage	1991
STI/DOC/10/322	Guidebook on Non-Destructive Examination of Water Reactor Fuel	1991
IAEA-Working Material NF-91/2	Analytical Report. Post-Accident Management of Destroyed Fuel from Chernobyl. Technologies Used and Lessons learned. A.A. Borovoi, Vienna May 1991	1991
STI/PUB/892	Nuclear Power, Nuclear Fuel Cycle and Waste Management, Status and Trends 1991	1991
IAEA-TECDOC-627	Catalogue of methods, tools and techniques for recovery from fuel damage events	1991
IAEA-Working Material NF-91/3	Micro-mechanisms of interface processes of materials degradation under irradiation: General Classification	1991
IAEA-Working Material NF-91/4	Survey of the available information on structural materials degradation in the nuclear fuel cycle back-end facilities in the relation to the IAEA CRP	1991
IAEA-Working Material NF-92/3	Spent Fuel Managmeent: Current Status and Prospects 1991	1992
IAEA-TECDOC-673	Extended storage of spent fuel. Final report of a Co-ordinated Research Programme on the Behaviour of Spent Fuel and Storage Facility Components during Long Term Storage (BEFAST-II) 1986-1991	1992
IAEA-TECDOC-679	Consolidation of spent fuel rods from LWRs	1993
IAEA-Working Material NF-93/1	Spent Fuel Treatment and Emerging Problems in this area	1993

OTHER RELATED ACTIVITIES IN THE IAEA

LONG TERM ISOLATED PLUTONIUM STORAGE

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Introduction

This article is intended to describe the circumstances which have led, and will lead, to significant accumulations of isolated plutonium. The intent is to draw attention to a developing issue which will require attention in the next 10 to 20 years. No attempt is made here to suggest how this problem should be dealt with or resolved - that will be the subject of future IAEA sponsored Advisory Group Meetings and Technical Committee Meetings.

Background

Early in the development of nuclear power, the demand for electricity was growing at about 7% per year - requiring installed capacity to double every 10 years. In addition, uranium supply was viewed as quite limited, at least relative to the projected growth in demand. Since this combination (limited supply and rapidly increasing demand) would rapidly drive the price of uranium up, an early and substantial capital commitment to fuel reprocessing and uranium recycling was made. This was the rational way to use very efficiently the uranium resources - in a once through fuel cycle, about 4% of the uranium is consumed; in a breeder recycle programme, nearly all of the uranium can be used.

However, after the initiation of construction of reprocessing facilities, the first of the "energy crises' occurred with the oil embargo . The increase in energy price and the greater awareness of energy issues seems to have permanently mitigated the rate of increase of electricity demand worldwide. But the combination of both the high capital cost of reprocessing facilities and the great "inertia" that commitment to their construction provided, has resulted in significant reprocessing capacity now on line. With the utilization of this reprocessing capacity, 10 - 15 tonnes of plutonium were isolated in 1990, and the rate of isolation of the plutonium is expected to grow to about 25 tonnes annually by 2000.

In addition to the significantly reduced electricity demand growth rate, two other factors have impacted the present and projected inventories of isolated plutonium.

First, the known uranium resources worldwide have increased significantly, thus reducing the pressure to find and develop alternate nuclear fuels. Second, there have been technical problems with the fast breeder reactor (FBR) so that the FBR development has been delayed in all countries. The effect of these two factors has been both to reduce the demand for isolated plutonium and to increase the uncertainty about future plutonium requirements. The existence and growth of the inventories of isolated plutonium have led to efforts to utilize it. The result of these efforts has been the development of MOX fuel for use in thermal reactors.

Economics of MOX Fuel

The evaluation of the economics of MOX as a nuclear fuel in thermal reactors is quite straightforward. One must compare the costs of a typical low enrichment uranium fuel cycle:

- the cost of natural uranium,
- plus the cost of sufficient enrichment services to produce the requisite low enrichment level for uranium fuel,
- plus the cost of fabrication of the low enriched natural uranium into fuel assemblies.

with the equivalent costs of a MOX fuel cycle:

- the cost of plutonium,
- plus the cost of (natural or depleted) uranium for matrix material,
- plus the cost of fabrication of the MOX fuel.

(To provide a complete cost comparison, the cost saving of final disposal of reprocessed waste instead of disposal of spent nuclear fuel from the once through cycle should also be included. However, since no final disposal has been completed for any fuel, these costs differences are difficult to estimate.) If the reprocessing costs are included as a component cost of the plutonium, most analysts conclude that the reprocessing of spent fuel and recycling the plutonium are significantly more costly than the once through (direct disposal) option. However, if the investment in spent fuel reprocessing and plutonium isolation is considered a "sunk cost" (an investment made long ago when the future of the fuel cycle appeared very different from the present reality - see assumptions above), the plutonium may be considered as a "free" material. This is because it is presently available and there are real costs of not utilizing it quickly (see below). In this case, the comparison of plutonium recycle in thermal reactors as MOX, with the typical uranium cycle, are closer in costs but still appear to favour the uranium cycle. This is because fabrication facilities for MOX fuel are more costly for three reasons. First, although reactor grade plutonium is not an ideal material for weapons, it can be so used. Therefore, MOX facilities must include security features to guard against weapon proliferation concerns. Second, because plutonium is quite toxic and has a relatively small critical mass, special safety features must be included. Thirdly, because of plutonium-241 decay to americium-241 which decays by gamma emission, more radiation protection for workers in MOX facilities must be provided (this problem continues to increase with increasing time between the reprocessing/isolating of the plutonium and fabrication of the plutonium into MOX fuel).

Conclusion

Because MOX fabrication facilities are more expensive, and the need for them is dependent on uncertain forecast economics and demand, investment in and development of such facilities is likely to be delayed. With the growing rate of isolation of plutonium and the very slow growing rate of use of plutonium, the inventories of isolated plutonium will continue to grow. Now, what should we do with the stuff?...

COUNTRY REPORTS

BELGIUM

1. GENERAL POLICY

Belgium has 7 nuclear power plants in operation, all PWR's with a total capacity of 5500 MWe.

In 1990 electricity generated by these plants reached 60 % of the total electricity production

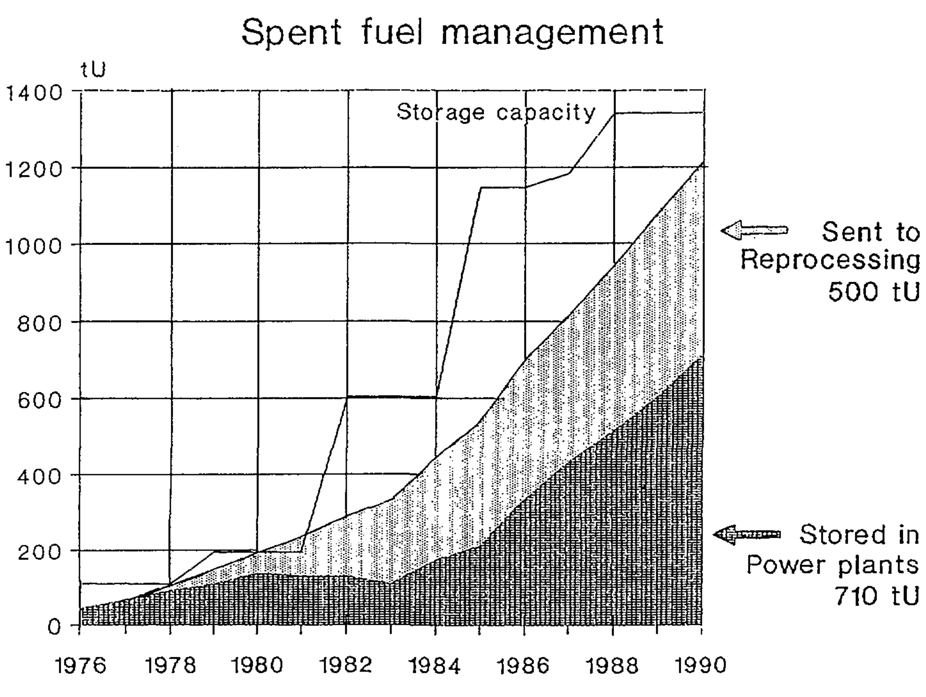
On the ground of conserving energy ressources, Belgium's general policy is to reprocess spent nuclear fuel, but delay between fuel unloading and reprocessing is depending on the legal, technical and economical environment.

2. SPENT FUEL STORAGE

Spent fuels are stored in the storage pools of the nuclear units before their transport to a reprocessing facility.

Evolution during the last 15 years of the spent fuel net storage capacity of the Belgian nuclear power plants (without taking into account a spare storage capacity corresponding for each unit to one whole core) and of the quantities of stored spent fuel is given in figure 1.

In the future, fuel storage capacity on site will be adapted in accordance with the quantities of fuel transported to reprocessing facilities in the frame of existing or new reprocessing contracts.



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3. REPROCESSING

Since 1976 four reprocessing contracts have been signed with the French company COGEMA for a total of about 670 tU with options and extensions. Quantities of spent fuel (140 tU) corresponding to the first 3 contracts have already been reprocessed by the end 1990.

4. URANIUM AND PLUTONIUM RECYCLING

The uranium recovered so far within the scope of the first three contracts has been sent to enrichment plants, whereas the plutonium has been used in priority for fast breeder reactors at Creys Malville and Kalkar.

Concerning uranium and plutonium recovered from 1990 to 2000 in the frame of the last contract :

- uranium will in principle be sent to enrichment plant and recycled the Belgian nuclear power plants
- plutonium will be used to manufacture MOX fuel for recycling in two nuclear power units.

5. WASTES

Conditioned wastes originating from the reprocessing of Belgian fuel will be returned in Belgium as from 1994 where they will be stored in storage facilities in construction at Dessel before their final disposal.

BRAZIL

Licensing and Control Superintendence
Nuclear Safety and Radioprotection Directorate
Brazilian Nuclear Energy Commission - CNEN

1. INTRODUCTION

Although the Brazilian Nuclear Programme envisaged the reprocessing of spent fuel, there is no effective government decision to build a plant for this purpose in the near future. However, in accordance with the federal legislation, the Regulatory Body (Brazilian Nuclear Energy Commission - CNEN) is in charge of the final disposal of the spent fuel and all other radioactive wastes.

There is in the country one nuclear power plant in operation. Angra 1, a 657 MW(e) PWR Westinghouse designed plant and there is another one, under construction, Angra 2, a 1300 MW(e) PWR KWU designed plant. The first research reactor, named IEA-R1, was built in 1956. It is a pool type reactor. Today, there are three research reactors and one critical assembly facility in operation.

2. Research Reactors

2.1 IEA-R1: IPEN/CNEN - Sao Paulo

The pool research reactor IEA-R1, 2000 kW(th), went into its first criticality in September 1957. The current fuel is a mixture of elements with uranium enriched at 93% and 20% in the isotope U-235.

The reactor pool is divided in two compartments and a gate may be placed between the operating and storage portions. In the storage compartment there are three special racks with 84 positions for spent fuel storage. The water conditions of the pool require a pH interval of 5.5 to 6.5 and a maximum conductivity of 2.0 micro mho/cm.

The following loads of fuel elements have been made:

1st. load: 40 fuel elements acquired from Babcok & Wilcox (USA), 20% enriched, curve plate, used during the commissioning of the reactor. In 1959, some fuel elements showed corrosion problems and consequently were replaced. This first load is in dry storage in the reactor building.

2nd. load: 40 fuel elements acquired from United Nuclear Corporation (USA), 20% enriched, curve plate. This load was used in the reactor core configuration up to 1976. The average burnup is 20% in U-235 and the spent fuel is stored in the reactor pool.

3rd. load: 40 fuel elements acquired from United States Nuclear (USA) in 1968 and 4 control fuel elements acquired from CERCA (France) in 1971, plane plate 93% enriched. From this load, 27 fuel elements with burnup rate higher than 30% are stored in the reactor pool as spent fuel. Seventeen fuel elements of this load are still in use in the reactor core configuration together with 5 fuel elements acquired from NUKEM (FRG) and 9 fuel elements manufactured in Brazil, both types 20% enriched in U-235.

2.2 IPR-R1: CDTN/CNEN - Belo Horizonte

The IPR-R1 research reactor, 100kW (the) TRIGA Mark I type, is running for 30 years with its initial fuel load, 58 fuel elements enriched at 20% in U-235,. 27 graphite elements, one central irradiation thimble, 2 pneumatic irradiation terminals, 3 control rods and one Ac-Be neutron source. Up to now no burned fuel has been unloaded. These fuel elements are on leasing and should be returned to the manufacturer (Gulf General Atomic).

Storage capacity for spent fuel elements on site is 2 racks with 6 fuel elements each, for temporary storage in the reactor pool. Inside the reactor hall, there are 12 concrete boxes under the floor, each one taking one rack with total available space for 72 fuel elements.

In order to acquire local know-how in burnup determination, an automatic gamma-scanning mechanism was designed and built and its main characteristics are:

- lead colimator of exchangeable aperture;
- longitudinal and rotational movements controlled by stepping motors;
- on-line data acquisition system;
- simplified biological (concrete plus iron) shield for decayed irradiated fuel elements.

2.3 ARGONAUTA: IEN/CNEN - Rio de Janeiro

The third research reactor, an ARGONAUT type, 200W(the) training category reactor, started operation in February 1965. The fissile material, enriched at 20% in U-235 come from the USA, but the fuel elements were manufactured in Brazil. It has a storage capacity of 24 underground cylindrical deposits located in the reactor hall. Until now no burned fuel has been unloaded. This reactor has two shielded casks to transport the fuel elements.

2.4 UCRI-MB01: IPEN/CNEN - Sao Paulo

This critical assembly facility, 100W(the), went into its first criticality in November 1988. Its fuel elements are 4.3% enriched in U-235. Until now no burned fuel has been unloaded.

3. Nuclear Power Plants

3.1 CNAAA-1: Furnas - Rio de Janeiro

The first Brazilian NPP, known as Angra 1,had its first provisional operating license issued in 1981. It has 121 fuel assemblies (28, 435 fuel rods), UO_2 sintered fuel pellets with maximum enrichment of 3.4% in U-235. The spent fuel is stored in racks which are composed of individual vertical cells that can be fastened together in any number to form a module that can be firmly bolted to anchors in the floor of the spent fuel storage area. The spent fuel storage racks are designed to include storage for 363 fuel assemblies. The spent fuel pit is a water-filled cavity designed to safety store three and one third cores of spent fuel and is constructed of reinforced concrete. Storage racks are erected on the pit floor and are designed with a center to center spacing of 16 inches. Adequate concrete and water shielding is provided to maintain radiation protection from the irradiated fuel assemblies. At present, (October 1991), there are 124 spent fuel elements stored. The Angra 1 Utility ("Furnas Centrais Electricas") is conducting studies for the design of super compact racks to replace the current racks in the spent fuel storage pit and the planned capacity is 1252 fuel assemblies by 1995.

3.2 CNAAA 2: Furnas - Rio de Janeiro

Angra 2 is a KWU designed plant, still under construction and it has a design capacity of the spent fuel pool for 772 fuel elements. The reactor core is designed to 193 fuel elements.

BULGARIA

INTRODUCTION

1. Nuclear facilities in Bulgaria

At the time being there is only one site with operating nuclear energy reactors,near the town of Kozloduy at the Danube river.There are six units in operation,four of them are WWER-440 type and two WWER-1000.No plans for construction of other units at the same site are foreseen.Construction of two WWER-1000s of the second bulgarian NPP-Belene started a couple of years ago.Due to a number of technical,financial, political and ecological reasons the Government took a decision to suspend the plant construction till the future nuclear energy policy is defined.

The only IRT-2000 research reactor-in Sofia was shut-down in 1989 for reconstruction.

2. Storage capacities

Kozloduy's spent fuel is to be stored for a three year period in water pools at reactors and after that to be transfered to AFRS or back to the supplier.Lately the supplier requires that spent fuel is stowed for a period of five years before it can be transferred back. This new requirement causes certain difficulties on the spent fuel management at Kozloduy.The storage capacity of WWER-440s is 360 FAs if fuel is stowed in one rack.It could be approximately doubled if second /upper/ rack is used.The design storage capacity of WWER-1000s is 440 FAs stowed in one rack only.

The AFRS/Away From Reactor Storage/is at the Kozloduy site as well.Transfer of spent fuel from at reactor ponds to it started three yers ago,but the operator has not yet obtained an official licence for comercial operation.33 baskets containing totally 990 FAs WWER-440 type were transfered there and some functional tests of cooling,ventilation and transportatin systems have been performed.Due to commissioning of WWER-1000s at Kozloduy the AFRS design has been slghtly changed to receive and store this type of FAs as well.Thus the total number of assemblies of both types stowed in it will be less than the designed-4900.in the Storage the FAs are stowed in baskets containing 30 items each.It is not clear yet which type of baskets for WWER-1000 fuel will be used.

The reasn for the delay in granting licence for AFRS operation is that the operator must fulfill some extra requirements enforsed by the regulatory authorities in compliance with the latest more stringent safety criteria.

Nowadays it is not foreseen to increase the AFRS capacity by using new type of baskets or racks.

SPENT FUEL ARISING

The annual theoretical WWER-440 spent fuel arising is 115 FAs per unit or 460 totaly.About 164FAs per year from the two WWER-1000s are expected to be included in the balance of the spent fuel at Kozloduy NPP.

TRANSPORTATIONS

According to the agreements the spent fuel is to be sent back to the supplier after a certain period of interim storage.Due to a number of new financial,technical and political circumstances in our relationships with the country-supplier no shipments took place over the last 3 years.This causes certain difficulties for our nuclear energy strategy and it is a problem to be solved with priority.

SAFETY

The main problem nowadays is the enforced storage of the spent fuel in two layers in reactor pools N^{o}2,3,4.The increased water level in the pools in that case leads to increased water leakage through the pools lining. Obviously the pools design and its performance are not up to the mark.

Another problem is lack of technology and equipment for spent fuel transfer out of the WWER-1000s containemets. This problem schould be solved till the end of 1992.

STORAGE

It is not planned at present to increase the WWER-440 ponds capacity by "rod consolidation" or reracking. There are no reprocesing plants or plans to construct such facilities for uranium and plutonium reprocessing as well.

CANADA

1.0 INTRODUCTION

The quantity of spent fuel being stored in Canada in water pools and in concrete, dry-storage canisters is approaching 800 000 CANDU fuel bundles (16 000 Mg U). Additional storage will be required at some sites in the 1990s, and will be provided by either additional water pools, dry storage in concrete containers, or some combination of these methods. The following is an update on spent-fuel management in Canada since IAEA Newsletter No. 1.

2.0 WATER POOL STORAGE

In Canada, all spent fuel from CANDU reactors is discharged directly into water pools at the reactor sites. A description of the pools and the movement of spent fuel between pools was provided in Newsletter No. 1. By the end of 1991, Canada will have a water-pool fuel-storage capacity for about 33 000 Mg U. At the present time Canada has 20 commercial CANDU pressurized heavy-water reactors in service, with a net capacity output of about 13 000 MWe. Two additional units, each having a net capacity output of 881 MWe, are either under construction or being commissioned. The typical annual fuel arisings from all of the above reactors is about 1 700 Mg U. The inventory of spent fuel bundles in water pools at the above reactor sites as of 1990 December 31 was about 14 000 Mg U.

2.1 Long-Term Fuel Integrity in Wet Storage

Ontario Hydro and Atomic Energy of Canada Limited (AECL) have a long-term program, initiated in 1977, to examine spent fuel stored in water for possible deterioration. The oldest bundles selected for the program have been in wet storage since 1962. The first examination under this program was performed in 1978/79, and a second ten years later. Results from the 1988/89 examination indicated no change in the condition of undefected CANDU fuel stored underwater for 27 years. The surfaces of UO_2 fuel fragments from intentionally defected spent fuel stored underwater for 21 years were highly oxidized and hydrated; however, the oxidation had no apparent effect on fuel element integrity. On the basis of the results from the most recent examination, it was concluded that spent CANDU fuel can be stored safely underwater for at least 50 years, provided that the rate of UO_2 oxidation does not increase significantly.

3.0 DRY STORAGE

3.1 AECL-Designed Concrete Canisters

In Canada, concrete canisters (CCs) are currently being used to store spent fuel at four sites: (1) AECL's Whiteshell Laboratories, (2) the Gentilly nuclear reactor site, (3) the Douglas Point Nuclear Generating Station, and (4) AECL's Chalk River Laboratories. Details of the concrete canister design were provided in the previous Newsletter.

Concrete canisters have been adopted as the storage method for future spent fuel discharged from New Brunswick Electric Power Commission's 633-MW(e) Point Lepreau Nuclear Generating Station. As of 1990 December 31, this station had completed almost eight years of operation, at a lifetime capacity factor of 90.4%. The Point Lepreau CCs have a significantly greater storage capacity (10.3 Mg U/CC) than the Douglas Point or Chalk River CCs (6.5 Mg U/CC), and have been licensed to store 6-year-cooled CANDU fuel in air, although only older fuel will be loaded initially. Of the 20 CCs constructed at Point Lepreau in 1990, 10 will be loaded in 1991 with spent fuel containing about 100 Mg U.

Since the previous Newsletter, spent fuel containing 65 Mg U has been loaded into CCs at AECL's Chalk River Laboratories. This fuel was discharged from NPD, Canada's first demonstration CANDU power reactor, which was shut down in 1987 after 25 years of operation. Also, another 1 Mg U of fuel from the WR-1 organic-cooled research reactor has been loaded into a CC at AECL's Whiteshell Laboratories. WR-1 was shut down in 1985 and almost all of its spent fuel is stored in CCs. By the end of 1991, fuel containing 556 Mg U will be stored in concrete canisters in Canada.

3.2 Ontario Hydro's Dry-Storage Container

In the autumn of 1990, Ontario Hydro loaded a second cylindrical demonstration concrete dry-storage container, with 384 bundles (containing 7.7 Mg U) of 6-year-cooled CANDU fuel from Pickering Nuclear Generating Station-A. Details of Ontario Hydro's Dry-Storage Container (DSC - previously referred to as a "Concrete Integrated Container" or "CIC") have been provided in the previous Newsletter. The demonstration container was loaded in the spent-fuel interim-storage pool, transferred by the pool crane to a trailer, then moved to an outside storage area. An upgraded rectangular DSC is currently in the design stage, to provide additional spent-fuel storage capacity at Pickering NGS. Preliminary drop and fire tests have been carried out on a 1/4-scale rectangular DSC for transportation development purposes. The successful 9-m and 1-m drop tests indicated that the container design could meet the impact test requirements of a Type B transportation package. The fire-test results indicated that neither the shielding nor containment integrity was compromised by the conditions required for Type B packages. For licensing purposes, the drop tests will be repeated on a 1/2-scale model, and an analytical model will be used to model a fire test on a full-scale container.

3.3 Long-Term Fuel Integrity in Dry Storage

In 1978, AECL and Ontario Hydro initiated a spent-fuel storage research program to investigate the long-term performance of spent fuel in dry storage. The experiments being conducted in support of the program are the Easily Retrievable Basket (ERB) experiment (dry storage of undefected spent CANDU fuel stored at seasonally varying temperatures) and the Controlled Environment Experiment, Phase 1 (CEX-1, storage of undefected and intentionally defected spent CANDU fuel in dry air at 150°C) and Phase 2

(CEX-2, storage of undefected, intentionally defected and in-reactor defected spent CANDU fuel in moisture-saturated air at 150°C).

Although a significant amount of fuel in intentionally-defected elements has experienced UO_2 oxidation in both CEX-1 and CEX-2, no evidence of fuel swelling has been observed, and fuel element integrity has been maintained. Significant oxygen depletion due to UO_2 oxidation and storage-container corrosion were observed in the storage containers holding defected fuel in both phases of the CEX. Since the amount of UO_2 oxidation experienced by the fuel could have been higher if the oxygen level had not been depleted, the experiment has been modified to allow the fuel to be exposed to an "unlimited" air supply. No change was observed in the condition of the fuel in any of the undefected elements examined since the fuel was characterized prior to storage.

4.0 DISPOSAL

In 1978, the governments of Canada and Ontario entered into an agreement to cooperate in the development of technologies for the safe, permanent disposal of Canada's spent fuel. The decision was made to focus research on the concept of disposal in plutonic rock in the Canadian Precambrian Shield, which is predominant over a large portion of Ontario. Under this agreement, Ontario Hydro is responsible for research on interim storage and transportation, while AECL is responsible for research on immobilization and disposal. In April 1981, the Canadian government approved, in principle, a ten-year generic research program.

In 1989 October, at the request of the Minister of Energy, Mines and Resources, an Environmental Assessment Panel was appointed by the Minister of the Environment to undertake a review of the spent fuel-disposal concept along with a broad range of spent-fuel management issues. The Panel review will be conducted under the requirements of the federal Environmental Assessment and Review Process (EARP).

The review process included public scoping meetings to identify the issues and assist the Panel in developing guidelines for the proponent (AECL) to prepare an Environmental Impact Statement (EIS). These scoping meetings were held in 16 cities in five provinces during the autumn of 1990. The draft EIS guidelines were sent out for comment in late 1991 June. The Panel will revise the guidelines in the light of comments it receives from interested publics, and publish the final version towards the end of this year. AECL is scheduled to submit the EIS in early 1993.

Once the EIS is submitted, it will be reviewed by the Panel and its Scientific Review Group to assess its adequacy, and the Panel may request AECL to provide further information. This will be followed by public hearings to review the proposal, and a report by the Panel in which it will address the safety and acceptability of the concept, will recommend what future steps should be taken. The Panel's report will be submitted to the Government who will then decide on the actions to be taken on the disposal concept.

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CHINA

1. Chinese nuclear power projects are being performing (See table 1). A total of nuclear power installed capacity will amount to about 6 GWe by the end of this century. After year 2000, the pace of developing nuclear power will be a little faster than that of the present century. It is predicted that the nuclear power installed capacity will come to around 30 GWe by the year 2015.

Table 1 China's nuclear power projects

power plant	Installed capacity MWe and type	Location	Startup
Qinshan 1 (prototype)	300 PWR	Haiyan, Zhejiang	1991
Qinshan (commercial)	300 PWR	Haiyan, Zhejiang	
Qianshan 2	600 X 2 PWR	Haiyan, Zhejiang	
Daya Bay 1 & 2	900 X 2 PWR	Shenzhen, GUANGDONG	1992 1993
Liaoning	1000 X 2 VVER	Liaoning	

2. The strategy of spent fuel management in P.R.China

According to the established national nuclear strategy, all spent fuel from NNPs ought to be reprocessed. After at least five years cooling in the reactor pools, spent fuel contained in casks will be transported to a reprocessing establishment by sea and rail.

Therefore, it is necessary to build a Central Spent Fuel Storage Facility(CSF) to provide reception and buffer storage of spent fuel to the reprocessing plants.

3. CSF

CSF will be located in the Lanzhou Nuclear Fuel Complex(LNFC) in north-west China. LNFC area is an ideal nuclear facility site where the population is very low and the meteorogical, geological and geographical conditions are all suitable.

CSF installations has a maximum reception capacity of 400 t HM/a with an overhead crane handling maximally a 100 tonnes grade transport cast.

The first storage pool(A) can hold 250 t HM in normal arrangement way and then 500 t HM if with compactness technique. The second identical storage pool(B) will be built to allow to double the capacity in the next stage. This capacity can be further expanded to 2000 t HM around the year 2010.

4. Reprocessing of spent fuel

A multipurpose reprocessing pilot plant with a capacity of 100 kg HM per day will be set up and put into hot operation late in 1990's and a large-scale commercial reprocessing plant with a capacity of 400 t HM per year will be set up around 2010 to meet futher commerical reprocessing requirement.

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CZECH REPUBLIC

1. Introduction

There are four PWR reactors in operation at Dukovany NPP (4x440 MWe, WWER type 213, first unit started operation in 1985) and two PWR reactors under the construction at Temelin NPP (2x1000 MWe, WWER type). There are also four research or training reactors in the territory of the Czech Republic.

The utilities are responsible for the safe managment of their spent fuel including necessary research and development as well as financing of spent fuel managment. Research institutes such as Nuclear Research Institute at Řež and universities participate in the research programme. The Czech Energy Board in Prague is responsible for the spent fuel and waste managment programme dealing with the spent fuel and wastes from NPP.

According to the Act No. 28/1984 Coll. any handling, or transportation of spent nuclear fuel in Czech Republic have to observe strict rules and all such operations are supervised by the state regulatory body. From January 1, 1993 the State Office for Nuclear Safety is the competent authority in Czech Republic, which shall supervise the safety of any operations with spent nuclear fuel. Each storage facility is also an object of inspections of IAEA, especially from the safeguards point of view.

The problems of the back-end of the fuel cycle were reviewed by different institutions and former regulatory body - Czechoslovak Atomic Energy Commission (CSAEC) understanding the situation decided that the utilities would have to prepare reasonable solutions (reprocessing, interim, storage, longterm storage, final disposal).

2. Spent fuel production and storage

Nuclear power plant at Dukovany represents the main source of spent nuclear fuel in Czech Republic. The fresh nuclear fuel has been supplied from the previous U.S.S.R. and was originally planned to be re-exported to that country. The transports of the spent nuclear fuel from Czechoslovak NPPs were stopped in 1988 and the spent nuclear fuel assemblies from NPP Dukovany were (after 3 years cooling period) transported to the wet interim storage facility at Jaslovské Bohunice in Slovak Repulic only. These transports were finished in 1992. There are about 1200 spent fuel assemblies from Dukovany NPP at the interim storage at Jaslovské Bohunice. This number of assemblies represents about 23% of the total capacity of this interim storage.

Owing to this situation the Dukovany NPP has decided to start building of dry interim storage at the Dukovany NPP site as well as to increase the capacity of the at reactor spent fuel ponds. The design of re-racking these ponds was approved by CSAEC. The compact racks will almost double the capacity of these at reactor spent fuel ponds, which will give the operator necessary time to build interim storage.

The interim storage will use the dry technology. It is supposed that the spent fuel assemblies will be stored in metal transport casks (Castor 440/84) which should be licensed for both storage and transport, respectively. The planned capacity of this interim storage is about 600 metric tons of heavy metal. The license procedure of the interim storage has not been finished yet.

The total amount of spent nuclear fuel assemblies of the reactors WWER 440 recently represents about 1400 assemblies in reactor ponds at Dukovany and about 1200 assemblies in the pools of wet interim storage facility at Jaslovské Bohunice in Slovak Republic.

Majority of the four research or training reactors - the reactor VR-1P at the Faculty of Nuclear Science and Physical Engineering in Prague, and LR-0 at the Nuclear Research Institute at Řež as well as the reactor ŠR-0 at Škoda Works Plzeň, produce no spent nuclear fuel, because of their zero power output. There is LWR-15 reactor operating at the Nuclear Research Institute at Řež which uses modern type of fuel U-AL alloy with the Highly Enriched Uranium. There are stored about 180 spent fuel assemblies IRT-2M type in the wet storage (pool) of LWR-15.

There is also a dry storage facility at the Nuclear Research Institute at Řež, where are stored almost 180 casks with approximately 3300 spent fuel elements of EK-10 type from the WWR-S reactor, which was reconstructed to LWR-15 in 1987. The fuel assemblies of the EK-10 type were enriched to 10% of ^{235}U, burn-up between 10 and 29% of the original content.

Both the wet and dry storages at Řež are rather old and a new building for both spent nuclear fuel and high radioactive wastes repository has been built and it is going to be commissioned. The designed building is situated within the area of Nuclear Research Institute.

The building is constructed as a one-cave hall the bottom part of which is built in the form of a monolithic basin from reinforced concrete that is divided into 12 cells. Seven cells are scheduled for radioactive wastes, 1 cell is equipped for storage of barrels with spent EK-10 fuel elements, 2 cells are designed as built-in storage pools for spent fuel elements of the IRT-M type one of the cells serves as an emergency reserve, 1 cell is reserved for technological water-service system and 1 cell is arranged as the entrance hall for transport conveyance.

3. Possible Solutions

The basic problem is caused by the spent fuel arising from the Dukovany NPP. The Czech Energy Enterprise is preparing several studies concerning this problem, but no final solution is prepared up to now. The following possible alternatives for Czech NPPs spent fuel managment exist :

* interim storage followed with the final disposal of spent fuel in Czech Republic
* foreign reprocessing and return of wastes to Czech Republic
* foreign final disposal services

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FINLAND

GENERAL

The Finnish power utilities operate four nuclear power plant units. Teollisuuden Voima Oy (TVO) owns at Olkiluoto, Eurajoki two units (BWRs, 2 x 710 MWe) and Imatran Voima Oy (IVO) at Loviisa two units (PWRs, 2 x 445 MWe). The units have been in operation since 1977 - 1980. The nuclear reactors produce about 30 % of the electricity in Finland.

The Nuclear Energy Act and Decree, passed in 1988, form the central regulatory basis for the management of spent fuel and other nuclear wastes. Responsibilities, licencing procedures and financing principles are defined in them. The objectives and schedules of nuclear waste management are set forth in the Government's policy decision of 1983 and in the stipulations given by the Ministry of Trade and Industry.

The utilities are responsible for the safe management of their spent fuel: research and development work, siting of the facilities, construction and operation of storage and final disposal facilities and financing of waste management. A number of research institutes, universities and consultants participate in the programme as contractors, such as the Technical Research Centre of Finland, the Geological Survey of Finland and the University of Helsinki.

The progress of the waste management programme is supervised by the Ministry of Trade and Industry. The Finnish Centre for Radiation and Nuclear Safety is responsible for the supervision of the safety of plans and activities. The major facilities are licenced by the Government. Part of the research work is funded by the Ministry in order to maintain independent expertise for the supervision of the programme of the waste producers.

SPENT FUEL ARISINGS AND STORAGE

The Loviisa power plant produce annually 28 tU (tons Uranium) of spent fuel. The spent fuel management of Loviisa is based on an agreement between IVO and the Soviet fuel supplier. According to this agreement, the spent fuel will be transported back to the Soviet Union after five years' storage at Loviisa. In the water pools at the Loviisa power plant, there is a total storage capacity for about ten years' need. By 1991, the total quantity of spent fuel produced is 350 tU.

The annual production at Olkiluoto is 45 tU. The total quantity produced by 1991 is 500 tU. TVO stores all the bundles at the Olkiluoto power plant site, in the storage pools of power plant units and of the interim storage facility (KPA store). The three water pools of the KPA store have a storage capacity of 1200 tU corresponding to the production of approximately 30 years. The store was commissioned in 1987.

FINAL DISPOSAL

Preparations have been done for domestic direct disposal of the Olkiluoto spent fuel since the end of 1970's. At the same time, the development of foreign commercial services have been followed.

The updated plans for encapsulation plant, canister and final repository were submitted to the authorities in 1990. The repository concept comprises horizontal tunnels with vertical holes in the floors at a depth of several hundred meters in the crystalline bedrock. For encapsulation, the cold process technique and the ACP canister (Advanced Cold Process) have been developed. The copper overpack of the copper-steel canister functions as a corrosion shield, and the steel support inside secures the mechanical strength. The final disposal facility is planned to be constructed in the 2010's and to be commissioned about 2020.

TRANSPORTATIONS

The spent fuel of Loviisa is transported to the Soviet Union by train in the Soviet wet flasks having a capacity of 30 bundles. By 1991, the quantity transported is 200 tU.

TVO has one flask for transfer of spent fuel bundles from the power plant units to the on-site store. The wet flask has a capacity of 41 assemblies. By 1991, 270 tU have been transferred to the store.

R & D WORK AND SITING OF THE REPOSITORY

According to the Government's decision, the site for final repository will be selected by the year 2000. Five granitic areas have been selected for preliminary site investigations. Field investigations were started at these areas in 1987. The programme consists of airborne surveys, deep and shallow drillings as well as measurements and sampling from the surface and in boreholes. Field work is followed by laboratory studies and modelling and evaluation of the areas.

In the autumn of 1990, TVO prepared a plan for studies of basic rocks in the site investigations. According to this plan required by the authorities, studies will be carried out during 1991-1993 to establish whether basic rock types differ significantly from the present granitic ones being investigated.

Parallel to bedrock investigations, the technology of encapsulation and final disposal is being developed and optimized. Various long-term experiments are carried out for safety assessments of the repository system. The performance assessments will be updated in 1992.

FINANCES

The Finnish utilities make financial provisions for the future costs of nuclear waste management. An updated cost estimate, including spent fuel, low- and intermediate-level wastes and decommissioning, has to be presented to the authorities annually. On the basis of these estimates, the Ministry of Trade and Industry each year confirms a fee to be paid to the State Nuclear Waste Management Fund. The fee is based on the estimated future costs for the management of the waste already produced including, however, also the costs of decommissioning.

The fee is adjusted in such a way that not later than after 25 years of plant operation the contributions paid into the Fund will cover all the future costs of waste management even if the plant would then be shut down and no more fees could be collected. Until then the power companies must furnish securities for

FRANCE

I - Introduction

French Nuclear Power plants supply about 75 % of the national electricity output. Therefore, spent fuel management is a fundamental activity. Since the beginning of its electronuclear program, France has opted for spent fuel reprocessing coupled with the recycling of recovered materials - uranium and plutonium - and with waste selective treatment and disposal. The processes and the technologies for reprocessing and waste treatment, developed in line with the other fuel cycle steps, have come now to a complete, mature industry with the operation of the UP3 reprocessing plant. Both french and foreign spent fuels, are reprocessed in french plants : to date, 29 foreign

electrical utilities have signed reprocessing contracts with COGEMA (9,000 tHM), specifying the return to the customer of wastes for final disposal.
This spent fuel management program is carried out in France, according to an appropriate structure including COGEMA, SGN, CEA and ANDRA.

2 - Spent fuel arisings

In 1990, three 1300 MWe PWR units have been connected to the grid and two GCR have been definitively shut down. At the end of 1990, the installed nuclear power capacity totalled up to 55,9 GWe with 56 units : 52 PWR, 2 GCR, 2 FBR.
The present core management stategy involves, for the 900 MWe units, an annual reloading scheme of 1/4 core and discharge burn-up of 36-42 GWd/t, instead of of a 1/3 core and burn-up of 33 GWd/t. At the end of 1990, 23 of the 34 units applied this strategy. In the future, higher burn-up in the range 50/60 GWd/t with annual reloadings 1/5 core might be used, leading to decrease the quantities of unloaded fuels.
In 1990, 916 tHM were discharged from the EdF PWR power plants (996 t in 1989).

3 - Spent fuel transport

COGEMA uses a specialized organization for managing all nuclear materials transports, and relies upon most competent international companies : Transnucléaire, Nuclear Transport Limited and Pacific Nuclear Transport Limited. COGEMA has now acquired a considerable industrial experience in spent fuel handling and transport. Over 3,000 transport casks shipments from 100 French, Western European and Japanese LWRs have been carried out so far, and over 11,000 tHM of fuel has been delivered by rail, by sea or by truck to the La Hague reprocessing plant since 1975.
For GCR fuels, 337 t of spent fuel were transported to Marcoule in 1990 (264 t in 1989). The cumulated quantity of GCR fuel transported so far is well above 9,000 t.

4 - Spent fuel storage

COGEMA's present storage capacity in La Hague is in the order of 10,000 t.

The dry storage facility CASCAD , chiefly designed for the fuel unloaded from the EL4 HWR, has been commissioned in 1990 at CADARACHE.

5 - Reprocessing

5.1 - GCR fuel reprocessing.
Until January 31,1987 GCR fuel elements have been reprocessed both in the UP1 (Marcoule) and the UP2 (La Hague) plants. For now on, UP1, first French reprocessing facility commissioned in 1958, remains the only reprocessing plant in France dedicated to GCR fuel. In 1990, UP1 reprocessed 310 tHM of GCR fuels. The cumulative tonnage of GCR fuel reprocessed in France since the beginning reached 8,900 t at the end of 1990 (4,900 t in UP2 and 4,000 t in UP1).

5.2 - LWR fuel reprocessing
COGEMA has now two industrial plants in operation :
. the UP2-400 plant, commissioned in 1966, adapted to LWR fuels since 1976, with the addition of an oxide head-end facility (HAO). From that time until February 1987, UP2 operated GCR fuel on alternating with LWR fuel. UP2-400 has then been entirely devoted to LWR fuel and has demonstraded an industrial capacity well above its nominal capacity of 400 t/y. New facilities have been added to the existing plant to form the UP2-800 project, designed to the same standards as UP3 for a capacity of 800 t/y. First industrial active operations will take place in early 1994.

. the UP3 plant, with a nominal capacity of 800 t/y. The UP3 plant start-up has been achieved in two successive steps. The first one, from November 89 to April 90, involved all facilities but T1, the head-end facility. During that period, shearing, dissolution and the first cycle extraction operations were performed in the UP2 plant. 100 tons of fuel have been reprocessed that way. The second step began in August 1990, involving the complete plant, with the T1 facility start-up. At the end of 1990, 125 t of fuel have been completely processed in the UP3 plant and 310 t at the end of July 1991, since its active start-up.

In 1990, 525 tHM have been reprocessed at La Hague. As of August 1991, the cumulative amount of LWR fuel reprocessed by COGEMA reaches about 3,900 t.

5.3 - FBR fuel reprocessing

At the end of 1990, about 28 tonnes of FBR fuel have been reprocessed : about 1 in the AT1 workshop at La Hague between 1969 and 1979, more than 10 t of fuel from Phenix between 1979 and 1984, in the UP2 COGEMA reprocessing plant at La Hague, about 17 t in the Marcoule Pilot Plant (APM) (about 11 t of fuel prior to 1984 and 5,7 t of Phenix fuel from 1988 to the end of 1990).

6 - Uranium and plutonium recycling

6.1 - Uranium recycling

France has gained, an important industrial experience in the conversion of the uranyl nitrate produced in the reprocessing plants :

- more thant 1 600 tonnes of reprocessed uranium (Rep U) have been converted into UF6 by COMURHEX in a dedicated demonstration facility started up in 1978, the capacity of which has been increased gradually from 30 to 350 t/year,

- significant quantities of recovered uranium have been transformed into oxide by COGEMA in its TU2 facility located at Pierrelatte. This facility, built in 1979-1980 has been recently adapted to produce U_3O_8 from reprocessed uranium at a rate of 600 t/year ; the same plant can also produce a sinterable UO_2 powder suitable for use as a MOX matrix.

Experimental fuel assemblies containing enriched reprocessed uranium, fabricated by FRAGEMA-FBFC, were loaded by EdF in one of its 900 MWe units (CRUAS 4)

6.2 - Plutonium recycling

Plutonium recycling as a mixed oxide fuel in FBRs has been carried out in France for a long time. More than 100 t of fuel have been produced in the CEA facilities at Cadarache (CFCa) since 1964 for the Rapsodie, Phenix and Superphenix fast breeder reactors.

In 1983, EdF decided to recycle plutonium in a program including at least 16 900 MWe units from 1995 onwards. MOX fuel is gradually phased in ; it was first loaded in November 1987 into the St Laurent B1 unit. Two other such operations were performed in 1988, three in 1989 and four in 1990.

The MELOX plant, devoted to MOX fuel manufacturing, with a nominal capacity of 120 t/y, is under construction at Marcoule by the MELOX Company (COGEMA and FRAGEMA), and should come in operation in 1995. Presently, the MOX fuel fabrication is carried out in Belgonucléaire's Dessel plant (35 t/y) and in COGEMA's CADARACHE facility (15 t/y).

7 - Waste management

7.1 - Waste treatment and conditioning

The French waste management policy is to minimize waste volumes and activities and to condition on-line the wastes from reprocessing in a solid form suitable for ultimate disposal. Thus, for the La Hague plants, the vast majority of activities HLW ends up in a small volume of glass, hulls and end pieces are embedded in cement, the sludges produced by the treatment of low and medium activity liquid wastes are incorporated in bitumen.

The R7 vitrification facility, in active operation since June 1989, is now vitrifying the backlog of fission products solutions generated in UP2, before to be associated to UP2-800. From hot commissioning to the end of 1990, 548 glass canisters have been produced from 540 m^3 of feed solution and 1,483 t of LWR fuel (830 canisters at the end of August 1991). Scheduled for starting early 1992, the T7 vitrification facility will vitrify fission products solutions from the UP3 plant.

The first French industrial vitrification facility, AVM, associated to UP1 plant has produced, from its June 1978 commissioning to the end of 1990, 1,841 glass canisters from 1,443 m^3 of fission product solutions.

The STE3 facility (liquid effluent treatment station), from its June 1989 commissioning to the end of July 1991, has produced 4,470 drums of bitumized wastes (1,838 drums in 1990). Bituminization has been used satisfactorily since 1986 at Marcoule in the UP1 plant where more than 50,000 drums have been produced.

7.2 - Waste disposal

The long-term industrial management of radioactive waste produced in France is the responsility of ANDRA.

7.2.1 -Management for short lived low level waste

The first French repository (Site de Stockage de la Manche-SSM) at La Hague commissioned in 1969, capacity 500,000 m^3, will be saturated and closed in 1994. A new site (Centre de Stockage de l'Aube-CSA) will be in operation this year. It will have a total capacity of 1,000,000 m^3. It is expected to receive around 30,000 m^3 of waste per year over a 30 year operating life.

7.2.2 -Management of long lived radioactive waste

One option retained in France for the long term management of long lived radioactive wastes (type B and C) is their isolation from the environment in a deep underground geological repository. In 1987, the Government requested ANDRA to start field investigations in 4 preselected sites with different rock formations : clay, granite, salt, schist.

Due to strong opposition at these sites, the Government decided to adjourn on field investigations February 1990 for a 12 months period. In the same time, two commissions were asked to make a peer review of the basic options concerning the management of the back-end of the fuel cycle. They published their reports respectively in December 1990 and February 1991. Last June, the French parliament approved the first draft of a new legislation covering a 15 year program for HLW deep geological disposal. This legislation will be debated at the Senate in October of this year. It also confirms France's new policy of exploring alternative waste conditioning before disposal, including actinide separation and transmutation. The final choice of a repository site would again be submitted to a parliamentary vote.

8 - Research and Development

Important R and D efforts have been carried out by the CEA in support of the spent fuel management industries : reprocessing, waste management, plutonium recycling.
To carry out these programs and the future ones, new installations named ATALANTE have been constructed at Marcoule. These new hot laboratories will be soon in active operation.

GERMANY

The basic principles for waste management in Germany are established in the Atomic Energy Act and in the waste management concept of the German Government which gives greater substance to the statutory requirements and the principles of the waste management provisions ("Entsorgungsvorsorge") for nuclear power plants. The resolution passed on 28 September 1979 by the heads of the State and Federal Governments confirms the integral waste management concept; it is be based in general on onsite and offsite interim storage followed by reprocessing of spent fuel, recovery of radioactive materials, and conditioning for the final disposal of radioactive waste. This waste management concept has continued to be put into practice in recent years. Significant progress has been made, but also some delays have been occurred.

The waste management concept embraces four significant steps:

I. Interim storage of spent fuel in the nuclear power plants and in offsite interim storage facilities.

II. Reprocessing of spent fuel and re-use of the nuclear fuel thus recovered in nuclear power plants (recycling).

III. Development of direct final disposal for spent fuel for which, in accordance with Article 9 of the Atomic Energy Act, reprocessing is technically not feasible or economically not viable.

IV. Disposal of radioactive waste in the stages
- Conditioning
- Interim storage in nuclear installations, in offsite stores in regional collection
- Intermediate storage of highly radioactive, heat generating waste (vitrified blocks) in interim storage facilities
- Final disposal

Secured provisions have been made for the accommodation of spent fuel from light water reactors - for six years in advance as demanded in the "Principles of waste management provisions for nuclear power plants" - primarily on the basis of interim storage within the nation's borders and reprocessing abroad.

1. The present situation

In Germany LWRs are in operation with an electrical output of 22300 MWe. The spent fuel arisings amount to 3600 tU so far 3000 tU have been transported to reprocessing facilities, 600 tU spent fuel assemblies are in the spent fuel pools in the reactors.

2. The interim storage capacity

The total storage capacity available is composed of the following: Onsite storage at nuclear power plants : approx. 5600 t are available in the old federal states and 250 t in the new federal states; 3000 t in offsite interim stores in the old federal states and 560 t in the new federal states, in total 3560t.

1500 t thereof in the Gorleben interim storage facility,
1500 t in the interim storage facility at Ahaus. The interim storage is in operation,
560 t thereof in the pool storage at Greifswald.

3. Reprocessing

Reprocessing of spent fuel from light water reactors and recycling of the recovered nuclear fuel in nuclear power plants are essential components of the waste management concept of Germany.

After termination of the German reprocessing plant, reprocessing services are contracted in France and UK. The available contracts and the consecutive options are covering the time period until 2015.

4. Conditioning, Interim Storage and Disposal for Radioactive Waste

Additional capacities for interim storage of radioactive waste were created with the opening of the store in Gorleben and the interim store in Mitterteich. In view of the available interim storage capacities for radioactive waste with negligible heat generation rates bottle necks may occur, if the Konrad final repository is not commissioned in time, since in 1996 - in some cases even earlier - the storage capacity of available or definitely planned facilities are exhausted.

The Gorleben repository is planned to receive all types of radioactive waste, particularly highly radioactive, heat-generating wastes; above-ground exploration of the salt deposit has been carried out.

Below ground exploration were initiated in September 1986 with the sinking of shaft No. 1.

The Morsleben repository in the new federal states is in operation since 1981. The safety situation of this repository will be reassessed to answer questions recently raised.Deposition of wastes is interrupted due to court decisions.

5. Direct Disposal

The feasibility of direct disposal of spent fuel was investigated as called for in the heads of government resolution of 28 September 1979; the safety evaluation was concluded on time.

On 6 May 1986 the German reprocessing company (DWK) submitted an application for a license pursuant to Article 7 of the Atomic Energy Act to construct and operate a pilot conditioning plant in Gorleben for the purposes of developing conditioning techniques for direct disposal.
In summer 1989, DWK's Board of Directors decided to construct the PKA, which is considered a necessary preliminary stage for the direct final disposal of spent fuel elements in Germany. Thus, the PKA will explore a supplementary back-end solution to reprocessing. The PKA construction is ongoing

6. Thermal Recycling of Pu

Germany decided to close the nuclear fuel cycle by reprocessing. The proposed recycling strategy uses LWR-MOX fuel assemblies for recycling plutonium. In 1980 the German utilities decided to cooperate with each other and with SIEMENS such that all plutonium is pooled and utilized for thermal recycling. The large-scale technical feasibility and the economic use of plutonium has thereby been demonstrated primarily for PWRs. The programme is extended to include BWRs as well in order to increase capacity available for recycling.

The experience gained from recycling of Plutonium is based on programs which to a large extent have been performed on a commercial basis. In total, about 100 000 MOX fuel rods containing up to 3.5 w/o Pufiss have been used in MOX fuel assemblies since 1966 with a total MOX-inventory of over 5 t..

7. International Cooperation

Germany cooperates through bilateral working agreements with other countries and through international organizations in the development of methods and processes for safe long-term storage and final disposal of radioactive waste.

Cooperation agreements covering activities ranging from regular visits and exchanges of experience to joint research projects exist with a number of countries.

Organizations and experts from Germany are playing leading roles in the planning and execution of extensive research and development programs within the European Community. An international exchange of information and experience takes place regularly within the Nuclear Energy Agency (NEA) of the OECD in Paris and the International Atomic Energy Agency (IAEA) in Vienna. Germany participates in a number of agreed research programs within the framework of the OECD. The Federal Government supports the work done by the IAEA in establishing rules and guide-lines in the field of waste management by sending of expert delegates.

HUNGARY

1. Paks Nuclear Power Plant

The Nuclear Power Plant is equipped with WWER-440 (V-213) type reactors. The four identical units took up commercial operation in 1983, 1984, 1986, 1987, respectively. The main technical data related to the spent fuel management are as follows:

- Fuel weight in each reactor	42 t
- Max. enrichment:	3,6 %
- No. of fuel assemblies in the core	349
- Refuelling interval:	once a year, one third of the core

In Paks NPP the spent fuel is stored under borated water in ponds located at each unit and connected via a sluice gate to the reactor pits. In the pond heat release and radioactivity of the spent fuel is reduced to values permitting transportation out of the facility.

Two independent technological systems provide cooling of the spent fuel pond water. Corrosion products of the cladding and of pond structural materials as well as other contaminants are removed from the system by water treatment. Due to the high purity of the water and the cleanness of the surfaces this water treatment system has to be run only for a few hours each month.

Should the activity of the primary circuit reach specified values during operation, fuel assemblies must be checked for leak-tightness upon removal from the reactor. Until now no inhermetical assemblies were found in Paks NPP.

Originally it was foreseen to keep the spent fuel at the site for three years before reshipping it to the supplier and so the design capacity of each spent fuel pond was 349 fuel assemblies.

Later on, storage time increased to five years, according to an agreement with the supplier and therefore the capacity of the spent fuel ponds had to be modified by reconstructing the internal rack structure for compact storage.

The fuel of WWER 440 reactors is of hexagonal cross section. Therefore the compact rack has a triangular lattice formed by hexagonal tubes. Subcriticality is ensured by geometry and by the rack material that contains 1,1% boron as neutron absorber. The present capacity of each pond provides 706 places (650 for hermetical and 56 for possibly inhermetical fuel assemblies).

Until now reshipment of spent fuel to the Soviet Union (now Russia) from NPP Paks took place 4 times after completion of five-year storage period on the site. Possibility of further reshipments is uncertain.

This problem can be solved with the establishment of an on site interim spent fuel storage. The NPP Paks selected the GEC-ALSTHOM Modular Vault Dry Store (MVDS). The contract was signed on 28 September 1992 for the design of the storage and licensing procedure will be carried out in this framework, too.

2. Research Reactor WWR-SM

Our first research reactor (then named WWR-S) started operations in March 1959 and run until 1967 at 2-2.5 MW thermal power level. It used fuel assemblies of EK-10 type. Each of them contained 128 g of U-235, enriched to 10 per cent. 82 fuel assemblies were discharged, with burnup reaching 25%.

A first reconstruction was performed in 1967. The new core consisted of 36% enrichment WWR-SM fuel assemblies, each containing 40 g of U-235 and it was surrounded by a metallic beryllium neutron reflector. This core produced a total of 11000 MWdays until May 1986. The average burn-up reached 50%. During this operational phase, 780 spent fuel assemblies were produced.

The reactor was again taken out of operation in 1986 for its second reconstruction. Now commissioning of the reconstructed reactor is underway. The authorized operational thermal power will be 10 MW, to be reached in fall 1993.

Two spent fuel storage ponds are attached to the reactor: The inner one is situtated in the main hall of the reactor. Its stainless steel tank is placed into a pit also provided with stainless steel lining, connected to the reactor tank via a duct. The grid arrangement of this pond is designed to ensure subcriticality under all conditions.

The inner storage pond can accomodate the spent fuel assemblies normally discharged during a two year period plus a total core. It has its own separate cooling system. Upper shielding is provided by 3 m of water and 30 cm of steel.

The external spent fuel pond is situated at a distance of 100 m from the reactor, on site. Its stainless steel tank, sunk into the soil, has a diameter of 2.5 m and is 7 m deep.

The fuel assemblies were placed in storage tubes. To prevent corrosion of fuel assmeblies during the storage period, the storage tubes are replaced now by hermetic containers of the same size. The containers are equipped with water and air sampling nozzles and their water can be drained and refilled.

Spent fuel is stored on site only temporarily, and the possibility of shipping back it to the supplier or an other solution to the storage problem (e.g. storage in the interim spent fuel storage to be built for NPP Paks) is under consideration.

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INDIA

Introduction

A distinguishing feature of the Indian nuclear programme has been its comprehensiveness, covering the entire fuel cycle; starting from uranium mining and milling, production of uranium, fuel fabrication, setting up research and nuclear power reactors, spent fuel reprocessing, recovery of plutonium and recycling and waste management.

India's nuclear energy programme took off with the commissioning of the first pool type research reactor `APSARA' in 1956. The first nuclear power station (BWR) consisting of 2 x 210 MWe reactors at Tarapur (TAPS) commenced operation in 1969. Since then, five more nuclear power reactors (PHWR) each of 235 MWe have become operational and one more similar unit is in the verge of commissioning. A number of 235 MWe and 500 MWe PHWRs are under construction or are being planned for execution to achieve a nuclear power target of 10,000 MWe in the near future. In order to make optimum utilisation of the country's limited uranium reserves and also to exploit the vast reserves of thorium, strategy of Indian nuclear power programme has been based on three stages, the first stage will have PHWRs using natural uranium as fuel which will produce plutonium for use along with thorium in the second stage through Fast Breeder Reactors (FBRs) and in the third stage U233 obtained from the second stage reactors would be used along with thorium to sustain a long term energy source. To meet this objective, the reprocessing option has been chosen for the spent fuel management programme in India.

The present status of the various steps of the spent fuel management programme can be summarised as follows:

Spent fuel arsings and storage

The twin ractor BWR station at Tarapur is in operation since 1969 with an annual discharge capability of about 25 tonnes of spent fuel at full power operation. The spent fuel arisings from this station is kept under storage. The storage pool at the reactor was initially designed to store 528 spent fuel assemblies. Subsequently the capacity was increased to 1500 spent fuel assemblies by reracking using high density racks. As this capacity also has been exhausted a separate wet storage facility Away From the Reactor (AFR) was constructed which has become recently operational. During the interim period when the AFR was under construction, spent fuel storage in dry casks was resorted to as an interim measure. In the spent fuel pool the decay heat from the stored spent fuel is continuously removed by recirculating the pool water through heat exchangers. The water quality and clarity are maintained by using precoat type filters with ion exchange resins (Powdex).

The AFR spent fuel storage facility has capacity for storage of 2000 spent fuel assemblies from TAPS. In this facility the spent fuel assemblies are stored in 12 x 12 arrays of high density fuel racks which are designed to maintain fuel in a geometry to limit K-eff less than 0.9 under conditions of optimum water moderation. Transportation of spent fuel from the reactor pool to the AFR pool is done using lead shielded casks which were designed for dry storage as well. These dry casks are designed for storage of spent fuel with a minimum cooling period of 10 years. The fuel pool cooling system is designed for the total load of 2000 fuel assemblies with a provision to increase it to 3312 assemblies.

RAPS 1 & 2 are the first two PHWR reactors to be constructed at Rajasthan. RAPS-1 was commissioned in 1973. The spent fuel generated from these reactors are stored in stainless steeel trays of the spent fuel pool at the station. Pool water cooling and purification system operates continuously to maintain temperature, water chemistry and activity within stipulated levels. The capacity of spent fuel storage pool is adequate for about 10 years full power operation of the two reactor units. The transportation of spent fuel from the reactor station to the reprocessing plant (located at Tarapur) is done by road and rail in 70 tonne casks in dry condition. The storage capacity of the pool at the reactor station has been increased by about 30% of its original capacity by reducing the spacing between each tray as also by increasing the stack height. It is also proposed to adopt dry storage in concrete casks for spent fuel with a cooling period more than 10 years. The proposed concrete cask will have 750 mm reinforced concrete thickness from all sides and 850 mm at bottom. Cask is lined with 6 mm thick steel plate both inside and outside and is designed to withstand mechanical stresses. The total weight of the cask fully loaded with spent fuel will be 60 Te. If necessary, AFR could also be used for storing the RAPS fuel.

All the twin reactor stations at Kalpakkam (MAPS), Narora (NAPS) and Kakrapar (KAPS) which are in operation or under advanced stage of completion are also provided with similar spent fuel storage capabilities.

Storage at reprocessing plant

Two reprocessing plants are under operation. The Trombay Plant is essentially for reprocessing spent fuel from research reactors. The PREFRE Plant at Tarapur is intended for reprocessing spent fuel from power reactors and has been reprocessing PHWR spent fuel. The method of storage of spent fuel at these reprocessing plants is wet storage under demineralised water. The Trombay plant pool has a storage capacity to store about 50 tonnes of spent fuel and the Tarapur plant has a capacity of about 45 tonnes. A minimum water shielding of 2.5 to 3 metres is provided above the spent fuel. There is no cooling provision in the reprocessing plant pools in view of low decay heat in the spent fuel stored due to sufficient initial cooling already undergone in the reactor pools. The pool water temperature is maintained below 42 degree C and pool water chemistry and clarity is maintained by the pool water clean up system consisting of cartridge filter, cation anion and mixed bed ion exchange columns. A pH range of 6.8 - 7.2 and conductivity below 0.1 micro S/cm are maintained in the pool. Regeneration of the ion exchange columns by nitric acid and routing the effluents to the waste evaporation system of the reprocessing plant has reduced the regenerant effluent discharge considerably.

The Reprocessing programme

The spent fuel management programme in India is mainly based on reprocessing option. The recovered plutonium is envisaged to be recycled in the thermal and fast breeder reactors. Reprocessing operations started in India in 1964 with the commissioning of Trombay Plutonium Plant for reprocessing spent fuel from CIRUS, a research reactor. Being the first plant, it was intended to demonstrate the indigenous capability and generate expertise in the field. The plant process cells were fully decommissioned and reconstructed with the expanded capacity to enable reprocessing of spent fuel from the Dhruva research reactor as well.

The first Power Reactor Fuel Reprocessing Plant (PREFRE) at Tarapur was built essentially to reprocess spent fuel from the power reactors TAPS 1 & 2 and RAPS 1 & 2. The plant has been operated under IAEA safeguards with PHWR spent fuel from RAPS.

Another reprocessing plant is under construction at Kalpakkam. This plant has provisions for making additions to enable reprocessing Fast Reactor fuels as well.

In view of the progressive increase in the spent fuel arisings from the planned growth of nuclear power generation in the country, reprocessing capacity is being enhanced in a phased manner to keep pace with the plutonium demands as well as to obviate the need to store large quantities of spent fuel in the storage pools. Medium size reprocessing plants are planned to be colocated at reactor sites with considerable spent fuel arisings so that large scale transportation of spent fuel across the country is avoided. A conceptual design of such a medium size reprocessing plant (350 te/yr) has been taken up recently.

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PAKISTAN

I. REACTORS

Pakistan has:

1) One 137MW(e) Nuclear Power Plant in operation at Karachi since 1971.

2) One 5 MW Swimming Pool Type Research Reactor (PARR-I) at Pakistan Institute of Nuclear Science & Technology (PINSTECH), Islamabad. This Reactor is being upgraded from 5 MW to 10 MW using LEU (20%) against original HEU (93%).

3) One Miniature Neutron Source Reactor (PARR-II) (27 KW) in operation since November, 1989.

All these facilities are under IAEA Safeguards.

II. REPROCESSING & SPENT FUEL MANAGEMENT

Pakistan signed a contract for construction of a Reprocessing Plant with a French company in 1973. A Safeguards Agreement with the IAEA was signed in March, 1976 (INFCIRC/239). In 1979 this company stopped work on the project. Since then all work/activity on this Reprocessing Plant has been suspended and the matter is subjudice.

The spent fuel from all the reactors is being stored in the reactor facilities.

RUSSIAN FEDERATION

1. NPPs with reactors of WWER-440, -1000, RBMK-1000, and BN-350, -600 types have been built and operated in Russia with yearly of spent nuclear fuel (SF). arisings of about 1300 t U.

The Chernobyl accident of 1986 caused cutting down of construction of NPPs with RBMK and WWER-440 reactors. Future nuclear program will be based on a new generation of power reactors meeting increased safety and reliability requirements [1, 2, 4, 5].

The major water-water concepts of the 3rd generation include: a high capacity WWER-1000 and two mean capacity (~ 630 MW(e)) projects. Commissioning of their demonstration units will start at 1998-2000 with simultaneous decommissioning of the 1st generation reactors.

By this time the AST-500 project shall also be realized (complete construction and commissioning) to solve problems of heat supply.

R & D studies on breeder projects based on BN reactor type as well as the construction of BN-800 reactors at the Uzhno-Ural site will continue to solve problems of closing U-Pu fuel cycle (with involving efficient burning of weapon Pu) and further improving the BN-reactor technology.

SF contains a significant quantity of fissibles forming a tangible contribution to nuclear raw material reserves. Therefore, SF can't be treated as mere radiactive wastes. By preliminary estimates, recycling of generated U and Pu in the fuel cycle of thermal reactors alone will halve the total demand in natural U during 2000-2030, and reduce capital expenditures for 1 GW(e) of installed capacity by 12-15%.

Today Russia (like all other countries adopting the closed fuel cycle policy) realizes the 1st step of the fuel cycle back-end. These activities involve SF reprocessing and use of règenerated U and partially Pu in thermal and fast reactors as well as storing vitrified radwastes of reprocessing process in above-ground facilities at the reprocessing plant sites.

SF of RBMK reactors makes an exception in that it will presumably not be reprocessed due to low fissible contents (U-235 - 0.4%; Pu-239+241 - 0.25%).

2. SF from WWER-440 and BN-350, and BN-600 reactors has been reprocessed since 1976 at the 1st home reprocessing plant RT-1 of 400+tU/y capacity near Chelyabinsk (enterprise "Mayak") [3]. The plant reprocesses all spent nuclear fuel from the former Soviet Republics and Eastern Europian socialist countries and Finland where NPPs have been built to Soviet designs and are using fuel fabricated at Russian plants. The final product of the plant is 2-2.4% U-235 enriched melt of hexahydrate of uranyl nitrate obtained by mixing reextract and highly enriched uranium solution and subsequent evaporation of the mixture. The melt is used for RBMK fuel fabrication. Pu in dioxide form is stored till the time of realizing the BN-800 concept.

SF fuel of WWER-1000 reactors will be reprocessed at a large capacity plant RT-2, which is now built near Krasnoyarsk.

The plant will be put into operation by turns of 1500 tU capacity each. The final product of the U-line is hexahydrate uranyl nitrate melt used for uranium reenrichment and fuel element fabrication. The product of the Pu-line is uranium and plutonium dioxide mixture which will be used for WWER-1000 and BN-800 fuel fabrication.

SF from WWER-1000 reactors is stored temporarily at the RT-2 plant site. This independently standing facility of 6000 tU capacity will be fully packed with SF by 2005. To date a storage capacity of 3000 t U has been already put to hot operation.

SF from RBMK reactors are stored in Independent storage facilities at the plant site for no less than 10 years. Researches are conducted into a concept of 30-year storage at the plant site or in regional facilities [6]. Filling the storage facilities with SF of the major reactor types at the 1st quater of 1992 is shown in the table [7]. In Russia SF is stored in water pools and this technology will dominate for many years to come.

Dry storage alternatives have also been developed, among these the most advanced are long-term storing of RBMK fuel in SS canisters placed in the canals of reinforced concrete blocks with decay heat removal by natural convection (reseach and designing stage), and short-term temporary storing of WWER-1000 SF in metal transportation casks TK-13 (demonstration stage).

3. In view of the substantial uranium reserves in Russia expansion of reprocessing capacities would not be economically attractive at least till 2005. Therefore, the home nuclear program during 1990-2000 with respect to fuel cycle back-end involves [8]:

- persisting the practice described in the previous section (S.2) as regards the management of all SF from existing WWER, BN and RBMK units;

- completing construction of a commercial complex for solidification of all types of liquid HLWs produced by RT-1 to reduce volumes of wastes stored in tanks. Constructing a commercial plant for extraction (during reprocessing process) of long-lived radionuclides to increase safety and reliability of waste management operations (by 1995);

- launching by steps construction of pilot plants for solid waste conditioning (since 1993);

- constructing an underground laboratory to develop technologies for final disposal of RBMK fuel and vitrified HLWs;

- choosing of the site and launching the construction of a repository for final disposal of RBMK fuel and vitrified HLWs in geological formations (2000-2010).

FILLING STORAGE FACILITIES WITH SF FROM MAJOR TYPES OF POWER REACTORS

Storage type / SF quantity	WWER-400		WWER-1000				RBMK			
	AR		AR		AFR		AR		AFR	
	Capacity	Filling	Capacity	Filling	Capacity	Filling	Capacity	Filling	Capacity	Filling
FAs	6000	3400	6100	2500	900	50	25000	21000	57700	42900
tU	700	400	2700	1100	400	22	2800	2400	6500	4900

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SLOVAK REPUBLIC

NUCLEAR ENERGY IN THE SLOVAK REPUBLIC.

The Slovak Power Enterprise (SEP) operates the nuclear power station at Jaslovske Bohunice, which is divided into two separate power plants:

the V-1 plant, with two VVER 440 (model 230) units, and
the V-2 plant, with two VVER 440 (model 213) units.

The first two units of the V-1 plant at Jaslovske Bohunice were connected to the grid in 1978 and 1980, and the two units of the second plant, V-2 in 1984 and 1985 respectively.

There are also four VVER 440 (model 213) units under construction at Mochovce. The first unit is supposed to start in 1994.

Other sites for nuclear power plants were selected in the past for further units, but these are now the subject of discussion about their suitability. It is not known, whether there will be any nuclear plants accepted or not.

BACK END OF THE FUEL CYCLE

The spent fuel assemblies are discharged into at-reactor pools. The fuel is stored in these pools for 3 years. After this period the assemblies are removed into the interim wet storage, which is situated at the NPP Bohunice. The interim storage was designed by Soviet designers. The total capacity of the interim storage facility is 168 baskets (this corresponds to 5040 fuel assemblies - ca.600 MTHM).

The original design assumed a 10 year period to store the spent fuel from the four units of NPP Bohunice. But at present this facility is also used for the storage of spent fuel from the Dukovany NPP (Czech Republic). After a storage facility is created by the Czech utility and the fuel is returned, the existing storage capacity at NPP Bohunice will be exhausted by the end of 1997.

The storage pools at the reactors of the Mochovce NPP will be compacted. This fact allows NPP Mochovce to store the spent fuel for a 6 year period. According to the start-up time schedule of Mochovce, there should be no problems with spent fuel storage till the end of 2001.

1. Long term storage of spent fuel

It is clear to us, that it will be necessary to store a lot of spent fuel assemblies from the VVER-440 reactors at the site of the plants. Presently there are 127 baskets with 3.696 fuel assemblies in the interim storage facility.

From the above-mentioned facts one can conclude that the total spent fuel storage capacity of the nuclear power plants in the Slovak Republic will be exhausted according to the following schedule:

- in 1997 at Bohunice,
- in 2001 at Mochovce

SEP will increase the storage capacity of the plants till the end of 1997 at the latest. The final decision, whether there will be an independent storage for each NPP or one common facility for both plants, will be made at the end of 1993. This new capacity must also fulfil the technical requirements for the new fuel which, will be developed in 1993 and 1994.

At present the average fuel assembly burnup is 32.000 MWd/tU and the maximum enrichment is 3.6%. It is expected that the new fuel can reach higher burnups in the range of 40-45 000 MWd/tU.

2. Final repository for spent fuel

Several preliminary geological studies were carried out to provide a basis for starting up the preparatory works which aimed to find the best solution for the final repository of spent nuclear fuel in Slovak territory.

It is expected that a final repository for the spent fuel will be constructed in close cooperation with foreign partners, who have a long time experience in developing concepts for the direct disposal of spent fuel.

3. Transport of the spent fuel

International fuel transport

At the beginning of the operation of the NPP Bohunice the spent fuel was transported to the USSR. These transports were stopped in 1988. Since then that have been no shipment back to the USSR.

From the beginning of 1992 there were some discussions with the experts from Russia about the transport of the spent fuel for eventual reprocessing.

Internal shipments to the away-from reactor storage facility

As mentioned above, the fuel after 3 years of storage in the reactor pools is transported to the interim storage facility at the NPP Bohunice. This transfer is done using the TK-30 rail transporter developed by the Fuel Institute in Freiberg (Germany). SEP owns two of these transporters. The containers have a fuel basket for 30 fuel assemblies, which is also used as a storage basket in the interim storage facility.

Prepared: M. Turner (SEP Bratislava)

SPAIN

In Spain two types of high-level wastes will have to be managed: the spent fuel from the nine light-water nuclear power reactors presently in operation, the most important type as regards quantity, and the vitrified waste arising as a result of reprocessing of the spent fuel from Vandellós I nuclear power plant, a graphite gas-cooled reactor which was definitively phased out in 1.990.

1.- SPENT FUEL AND HIGH-LEVEL WASTE GENERATION FORECAST

Present estimates are based on an installed nuclear power capacity of 7.1 Gwe and on a plant lifetime of 30 years, the refuelling outages and cycles foreseen by the plants themselves taken into account. Total spent fuel and high-level waste arisings are included in Table below; volumes may be subject to changes, depending on the conditioning finally adopted.

HIGH LEVEL WASTES	M3
LWR Spent fuel	11500
5224 tU made up of:	
. 9321 PWR fuel elements	8520
. 6468 BWR fuel elements	2980
Vitrified wastes (Vandellós I)	180
	11680

2.- SPENT FUEL INTERIM STORAGE

At present, storage of the light-water reactors spent fuel is taken place at the pools of the nuclear power plants. As a period of some 30 to 40 years may be necessary before its final disposal, additional storage capacity will be needed. Current strategy contemplates to provide additional capacity at each reactor site, either by means of increasing the capacity of existing pools or by using dry storage in metal casks, as well as to build in due time a centralized interim storage facility according to foreseen needs.

The former solution is presently been provided at some nuclear power plants by means of reracking and, at the same time, the development of dual-purpose metal casks (transport/storage), which could be used both at-the-reactor-site and away-from-the reactor site, has been undertaken.

Construction of a centralized interim storage facility, for which a preliminary non-site specific project was already performed contemplating a combination of wet storage (pools) and dry storage (metal casks), is being evaluated.

In tables below, a breakdown of spent fuel generation forecast, as well as the current storage available and foreseen capacities are provided.

ACCUMULATED SPENT FUEL GENERATION FORECAST (THM)

	up to 1990	1995	2000
LWR	974	1770	2625
GCR	445	447 (x)	-

(x) Last core from Vandellós I nuclear power plant.

SPENT FUEL STORAGE CAPACITY (*) (THM)

1990	1995	2000
1950	4100	4250

(*) Full core discharge capacity not included.

3.- FINAL DISPOSAL

Along with the development of interim storage systems for spent nuclear fuel a deep geological disposal programme is in progress. Three main lines of action are contemplated in such a programme: Siting process, repository desing and associated R + D plan. Some projects are currently being developed in each line.

Background

The nuclear power programme

Sweden has twelve nuclear power reactors (3 PWRs and 9 BWRs) with a combined capacity of 9 900MW net electric power. They generate about 66 TWh (≈ 7.8 MWh per capita) annually which corresponds to about 50% of the total electricity consumption.

According to a resolution passed by parliament in 1980, Sweden will terminate its use of nuclear power in the year 2010, at the latest. Thus, an assessment can be made of the total volumes of spent nuclear fuel and other types of waste generated in this programme.

Waste management guidelines

According to generally accepted guidelines, the spent nuclear fuel will be kept in interim storage for a approximately 40 years after which, according to present plans, it will be deposited in geological formations in Sweden. No reprocessing will be performed.

The interim storage for the spent fuel will provide for considerable decay of the radioactivity and consequent reduction of the heat generation. This will facilitate the design and construction of a final repository. Another advantage of the interim storage is that it leaves ample time for the R&D work needed.

Responsibilities

According to Swedish law, the primary responsibility - technically and financially - for the safe handling and disposal of the radioactive waste rests with the owners of the nuclear power plants. In compliance with this requirement, the nuclear power companies have commissioned their jointly owned Swedish Nuclear Fuel and Waste Management Company (SKB) to develop, construct and operate facilities for spent fuel storage and waste disposal.

In order to fulfil this task, SKB is conducting a comprehensive research, development and demonstration programme for the disposal of spent fuel and is operating the Interim Spent Fuel Storage Facility (CLAB) and the Swedish Final Repository for Radioactive Waste (SFR) which is designed to accommodate short-lived waste.

Moreover, a principle has been established that the total costs for the waste management shall be borne by those who benefit from the electricity generated by nuclear power. Thus - also in accordance with Swedish law - fees are paid on the electricity and the money is set aside in interest-bearing accounts. The fees, together with the interest accumulated, are estimated to cover all costs - now and in the future - for handling and disposal of high-level waste and for decommissioning of the nuclear power plants.

The nuclear activities of the industry in Sweden are overseen by three government agencies: *the Swedish Radiation Protection Institute (SSI), the Nuclear Power Inspectorate (SKI) and the National Board for Spent Nuclear Fuel (SKN).*

SSI oversees the industry's activities with regard to protection of the employees as well as of the public against radiation hazards. SKI oversees the safety of nuclear power plants and any other facility where nuclear waste and fissile material is handled. SKN evaluates and supervises SKB's R&D programme on the management and disposal of spent nuclear fuel and the decommissioning of nuclear power plants. SKN also administers the system for financing and is responsible for making impartial information on nuclear waste matters available to the public.

Interim storage

The currently estimated total quantities of spent nuclear fuel to be generated in the Swedish programme is about 7 800 tonnes.

Since 1985, the spent fuel is stored in the *Swedish Interim Spent Fuel Storage Facility (CLAB)* located at the site of the Oskarshamn nuclear power plant. The storage is of the wet type comprising large water pools located in crystalline rock about 25 meters below the surface. The store has a present capacity of 3 000 tons of spent fuel (counted as uranium metal) and at the end of 1990 the store contained 1 350 tons.

The plant has operated according to plans and the performance has been good. The occupational doses as well as the releases have all been well below the regulatory limits.

SKB has found that the storage capacity can be increased to 5 000 tons if new compact storage cassettes made of the neutron absorbing material borated steel are used. In late 1989, the government gave its consent for the use this method. The compact cassettes are expected to be licensed and in use by March 1992.

Transportation

Since all nuclear reactors in Sweden as well as CLAB and SFR are located at the coast with good harbour facilities, a transportation system based on a sea vessel and terminal vehicles has been developed.

The system consists of the specially designed ship called *M/S Sigyn*, 10 transport casks for spent fuel, 2 transport casks for core components, 27 IP-2 type containers (ATB) for transport of low- and intermediate level waste and 5 special vehicles for loading and unloading as well as transfer at the terminals of casks and containers.

The shipments of spent fuel and wastes from reactor operation have been carried out in accordance with the plans and without disturbances. The doses to the crew have been below the detection limit.

Disposal of high-level waste

SKB R&D Programme 89

As required by Swedish law, SKB submitted its second R&D programme to SKN for review in September 1989.

The major elements of this plan are as follows. 1991-2000: construction and operation of a hard rock test laboratory, system selection, site selection. 2001-2010: siting application, preliminary safety report, SSI and SKI review, final safety report. 2011-2020: construction and commissioning of encapsulation plant and final repository.

SKNs evaluation of the R&D Programme 89 was based on comments from a number of other government agencies (including SSI and SKI), organisations and interest groups. Several recommendations were made eg that SKB should investigate whether the disposal could be achieved in stages which would make it possible to re-evaluate the situation at the end of each stage.

In its decision, the government ruled that, in general, SKB should adopt or respond to the advice given by SKN. The government also expressed as its opinion that very deep boreholes and very long tunnels under the seabed of the Baltic (suggested by SKB as alternatives to the KBS-3 concept) appear to be less suitable as disposal systems.

The Äspö Hard Rock Laboratory

The construction of the *Äspö Hard Rock Laboratory (HRL)* was initiated in October 1990. At present (September 1991), 900 meters of tunnel have been excavated to a depth of 130 meters. The main excavations are expected to be finished in July 1994 but several experiments will be carried out in parallel with the construction work.

The objective of the HRL is to test which methods that are most appropriate for investigating the bedrock, to refine and demonstrate methods for how to adapt a final repository to the local properties of the rock and to collect material and data of importance for assessing the safety of the final repository.

New safety assessment studies

The latest comprehensive description of a possible Swedish disposal system for spent fuel, called KBS-3, appeared in the safety analysis by SKB in 1983. This is still the reference concept of SKB although also other alternatives are studied.

According to the KBS-3 concept, the spent fuel will be enclosed in thick copper canisters which are to be deposited into holes drilled in the floors of tunnels with compacted bentonite as buffer and backfill. The tunnels will be excavated in good quality crystalline rock at a depth of about 500 meters.

In August 1991, SKI presented the results of *Project-90*. The project was initiated in order for SKI to acquire a basis for formulating guidelines and to prepare itself for future reviews of license applications. Project-90 is a performance assessment study of a reference repository with basic characteristics from the KBS-3 concept and with a synthetic reference site. The work comprises an identification and characterization of scenarios as well as a sensitivity study of the release and transport of radionuclides.

At the end of 1991, SKB is expected to report its new safety assessment study called *SKB 91*. The study is in particular aimed at illustrating the significance of rock properties for the total safety of the system. In the study, a KBS-3-like repository is placed at a site having a rather high hydraulic conductivity in the bedrock.

International co-operation

Much of the work conducted by SKB and by the government agencies SKI, SKN and SSI is carried out in an international context. SKB, in particular, has many agreements on international co-operations.

Only a few examples of this international work is mentioned here.

Under the sponsorship of OECD/NEA, seven countries are co-operating in the *Stripa Phase 3 project*, lead by SKB. The experimental part of the program is now completed and work is in progress to compile and evaluate the results. There are two main areas of work:

- Fracture flow and nuclide transport with site characterization and validation as the major subproject.
- Ground water flow path sealing using grouting techniques.

The Stripa mine is now filled with water and future hard rock laboratory work will be carried out at the Äspö facility.

The *Äspö Hard Rock Laboratory (HRL)* mentioned above is attracting international interest. At present (October 1991) SKB has signed agreements for co-operation with CRIEPI and PNC in Japan and AECL in Canada.

Five countries have participated in the natural analogues project called *Poços de Caldas* which is now closed. Sweden was represented by SKB.

Since April 1989 AECL (in Canada) and SKB are engaged in the *Cigar Lake* natural analogue project and the results of the first phase were reported at a workshop in Pinnawa in April 1990. The main objective of the second (ongoing) phase is to describe and model the water-mineral interactions in the uranium deposit and the trace element migration around it.

A new Nordic three year R&D programme on radioactive waste and decommissioning was started in 1990. Of particular interest in the present context are the studies of *geological and climatological processes of importance for long term repository performance* and of *information conservation* which are actively supported from the Swedish side by SKB, SKI, SKN and SSI.

SKI, SSI and the nuclear safety and radiation protection authorities in the other Nordic countries have published a joint consultative document on *criteria for final disposal of high-level radioactive waste.* Another similar document has recently appeared as a result of collaboration between the Swedish authorities and HSK in Switzerland.

Much of the work within the SKI Project-90 mentioned above was carried out in an international context.

SKI has also managed the international project *INTRAVAL* which closed its first phase in 1990. The purpose of the study was to increase the understanding of how various phenomena of importance for the transport of radionuclides from a repository can be described by models.

SKI has also initiated the co-operative project *DECOVALEX* which held its first meeting in early 1991. The purpose of this project is to develop models that describe coupled thermo-hydro-mechanical processes that are potentially important for determining eg repository induced alterations of the rock mass as well as faulting and other large scale phenomena.

BIOMOVS (BIOsphere Model Validation Study) is an international co-operative effort to test biospheric models designed to calculate the environmental transfer and bioaccumulation of radionuclides and other trace substances. The first phase of BIOMOVS - which was managed by SSI - was concluded in Stockholm 1990. A second phase was initiated at a meeting in Madrid in February 1991.

In february 1991, SKN signed a letter of understanding with the United States Technical Review Board. So far co-operation has been initiated in the areas of *canister materials and excavation methods.*

UNITED KINGDOM

Background and General Issues

1. The UK's nuclear generating capacity comprises some 3200 MW Magnox and 5100MW AGR operated by Nuclear Electric (NE) and 2450MW AGR operated by Scottish Nuclear (SNL) together with 400MW Magnox operated by British Nuclear Fuels plc (BNFL). The only additional nuclear capacity currently under construction is the 1175MW Sizewell B PWR owned by NE and which is expected to be in operation in 1994.

2. Further expansion of the UK's nuclear generating capacity will depend on the outcome of a Government prosed review in 1994. Although planning permission for the proposed Hinkley Point C PWR station was approved i September 1990, construction approval will hinge on the Government's 1994 review. NE are currently indicating a preference for a twin reactor station at Sizewell (Sizewell C) as the next stage of its development programme. All of the Nuclear Generating Companies in the UK have a controlling Government shareholding.

3. The main events that have taken place since the 1991 status report to the Advisory Group are as follows:

3.1 Towards the end of 1992 a House of Commons Trade and Industry Select Committee was set up to consider the consequences of British Coal's pit closure programme for the electricity consumer, the Exchequer and the economy and to examine alternatives in terms of energy policy. NE, SNL and BNFL have all given evidence to this Committee in respect of nuclear power generation. At the same time an internal Department of Trade and Industry Energy Review has also been underway. The findings of these two reviews are expected to be announced early in 1993.

These proceedings will undoubtedly have a bearing on the 1994 Government Review of the nuclear industry. However, the extent to which they may determine the scope and significance of the review is not yet known.

3.2 In the first two years of trading following the restructuring of the UK's electricity supply industry both NE and SNL have declared increasing profits and turnovers. In the 1991/92 financial year NE declared an operating profit of £ 482M on a turnover of £ 2,432M whilst SNL declared an operating profit of £ 68.7M on a £ 477M turnover.

3.3 Nirex are currently undertaking detailed site investigations to confirm Sellafield as the location of the proposed ILW/LLW repository. The additional costs of transport of ILW from Sellafield (where the bulk of ILW is produced) effectively ruled out the alternative candidate site at Dounreay. As part of these investigations Nirex intend to develop an underground Rock Characterisation Facility during 1994, subject to planning approval. The planning application for the repository itself is expected to be the subject of a Public Inquiry, likely to take place in the mid 1992. Operation of the repository is anticipated to commence towards the end of the first decade of the next century.

3.4 During 1992 the construction of the THORP plant, at Sellafield, was completed and work was concentrated on commissioning activities.s Active commissioning is expected to commence later than originally anticipated. This is due to a delay in the start of the 8 week public consultation period which forms a necessary part of the process which will lead to the approval of new Sellafield Site Discharge Authorizations. These authorizations will allow active commissioning to begin. Her Majesty's Inspectorate of Pollution announced on 16th November 1992 that the public consultation period had begun. This was completed on 25th January after a two week extension.

SPENT FUEL MANAGEMENT

Magnox Fuel

4. The strategy on Magnox fuel remains unchanged since the 1991 status report. Prompt reprocessing is essential and the technical reasons for this have already been rehearsed with the Advisory Group. All fuel will continue to be dispatched to BNFL's reprocessing facility at Sellafield.

AGR Fuel

5. Agreement has been reached in principle between BNFL and SNL for the reprocessing in THORP of a further 330t U beyond their existing commitment of some 600tU. SNL are currently pursuing the alternative of dry storage followed by either direct disposal or deferred reprocessing for the remainder of their AGR arisings. On 1st December 1992 a Public Inquiry began into SNL's proposal to construct a dry storage for AGR fuel at their Torness power station. The Inquiry ended on 19 January and the outcome is expected to be announced within some months. Beyond current agreements NE has made no decision regarding the spent fuel management strategy it will adopt for later arisings of AGR fuel.
6. In all other respects the position on AGR fuel remains as reported in 1991.

PWR Fuel

7. Strategy remains unchanged. Although indicative costs for spent fuel management were provided to both the Sizewell B and Hinkley C public inquiries in terms of reprocessing, the reprocessing and direct disposal options remain open. The current UK design of WR provides for 18 years of at-reactor wet storage and this would increase with the adoption of higher burn-up and/or rod consolidation. Full use will be made of the storage capacity available at the reactor. Decisions on reprocessing or direct disposal do not have to be taken for some considerable time.

Recycle of Uranium

8. Over 15000 tU of the uranium recovered by Magnox reprocessing has been recycled and a large proportion of all AGR fuel has been produced from this material. The recycle of reprocessed uranium from Magnox and AGR fuel currently has no strategic benefit and is assessed against alternative commercial options. The forecast uranium market conditions are such that further MDU recycle is unlikely and this is also likely to influence recycle of ORP. BNFL have however made proposals to NE and to SNL for trial loadings in AGR in the next few years of recycled uranium with characteristics similar to those expected in uranium with to those expected in uranium recovered from THORP operations.

Recycle of Plutonium

9. The position remains unchanged from that previously reported. Briefly, recycle to fast reactors remains the preferred option but, given the Governments recent decision on the fast reactor, in the shorter term recycle to possible UK PWRs including Sizewell B is an option. Recycle to AGRs currently appears unlikely on economic grounds but all options remain under review.

Fast Reactor

10. The UK Government has signalled its withdrawal of support for the Dounreay Fast Reactor project in 1994 and has also ruled that the nuclear industry in the UK cannot itself fund the project. Government support for the European Fast Reactor project will also be withdrawn in March 1993.

XA0102658

UNITED STATES OF AMERICA

The inventory of spent fuel in storage at reactor sites in the United States at the end of 1992 was approximately 21,000 metric tons heavy metal (MTHM). It is increasing at a rate of 1700 to 2600 MTHM per year. According to current projections, by the time the last license for the current generation of nuclear reactors expires, there will be an estimated total of 84,000 MTHM. No commercial reprocessing capacity exists or is planned in the U.S. Therefore, the continued storage of spent fuel is required. The majority of spent fuel remains in the spent fuel pools of the utilities that generated it. Five utilities are presently supplementing pool capacity with on-site dry storage technologies or utilizing dry storage after the reactor is no longer operating, and several others are planning dry storage. Commercial utilities are responsible for managing their spent fuel until the Federal waste management system, now under development, accepts spent fuel for storage and disposal.

Federal legislation charges the Office of Civilian Radioactive Waste Management (OCRWM) within the U.S. Department of Energy (DOE) with responsibility for developing a system to permanently dispose of spent fuel and high-level radioactive waste in a manner that protects the health and safety of the public and preserves the quality of the environment. We are developing a waste management system consisting of three components: a mined geologic repository, with a projected start date of 2010; a Monitored Retrievable Storage Facility (MRS), scheduled to begin spent fuel acceptance in 1998; and a transportation system to support MRS and repository operations. The following is an update of the current status and plans for the repository, MRS, and transportation elements of the program.

<u>Repository</u>

The repository will include three components: the natural system, the mined repository, and the waste package. The natural system-or site-consists of the host rock and surrounding rock formations. The repository consists of various underground structures and components, including tunnels and disposal rooms. The waste package consists of the waste form, the disposal container, and any material or feature separating the package from the rock. Currently, we are investigating the Yucca Mountain site in Nevada to determine its suitability for repository development.

Yucca Mountain is located about 100 miles northwest of Las Vegas, Nevada, on lands controlled by the Federal government, including the Nevada Test Site. The candidate site is located in an arid region with sparse vegetation and few inhabitants. The rock is volcanic tuff, and the proposed repository would be constructed in the unsaturated zone above the water table. The Yucca Mountain site has not been selected for a repository. Rather, it has been designated by the U.S. Congress as the candidate site to be characterized. DOE is conducting a program of detailed scientific investigations and suitability evaluations at the Yucca Mountain site to determine whether it is suitable for development as a high-level radioactive nuclear waste repository. The plans, activities, and results of this program will be reviewed by the State of Nevada, Nuclear Reglatory Commission (NRC), Nuclear Waste Technical Review Board (NWTRB), and other external organizations.

In the near term, DOE's scientific investigations have been focused on any features that would indicate that the Yucca Mountain site is not suitable for a repository. This site characterization program includes surface-based testing, and investigations conducted in an Exploratory Studies Facility (ESF) constructed to provide access to the underground rock formation in which the repository would be built. DOE will further develop the designs for the repository and waste package as more information becomes available about the suitability of the site.

In December 1991, OCRWM received approval to start comprehensive surface-based testing and to initiate the Final Design for the ESF. Ground breaking for the construction of the north portal and a 200 foot launch chamber for the ESF was held on November 30, 1992.

MONITORED RETRIEVABLE STORAGE

The MRS will be an above-ground, engineering facility that will receive shipments of spent fuel from commercial utilities, temporarily store the spent fuel, and stage shipments of spent fuel to the repository.

The Nuclear Waste Policy Act, as ammended (NWPA), authorizes two paths for MRS siting: 1) siting by the DOE, through a process of survey, evaluation, and selection, and 2) siting through the efforts of the Nuclear Waste Negotiator. Our present strategy is based on siting through the Negotiator, but we are developing a contingency siting plan and will decide on the basis of the MRS schedule and status of external efforts as to the appropriateness of implementing that contingency plan. Our target date for spent fuel acceptance at an MRS is 1998; therefore, an MRS host needs to be identified as soon as possible. The MRS will operate for approximately forty years.

In developing a proposed agreement with the Negotiator, a host State or Indian Tribe can negotiate for itself an active role in MRS development and operations. By participating in decisionmaking and by exercising rigorous oversight of MRS activities, the host can assure itself that the MRS will perform to its satisfaction, meet community standards, and serve community goals. Several Indian tribes and local communities have expressed an interest in hosting the facility.

We are supporting the Negotiator's efforts and preparing for development of the MRS. Preparations include development of potential MRS configuration and design options, planning for conducting an environmental assessment (EA) of a potential MRS site, and planning for MRS licensing.

We have recently completed the conceptual design for the MRS facility. The purpose of the conceptual design is to demonstrate that the MRS is technically feasible, to establish technical performance levels, and to develop reliable cost estimates and realistic schedules. The design was developed by carefully examining the requirements that the MRS must meet--including compliance with safety and environmental regulations, licensing requirements, and compatibility with the repository and transportation systems.

The MRS will include areas for receiving, handling, packaging and storage of spent fuel; and for support and industrial services. Depending on the storage concept that is chosen, the storage area may be a large paved-over yard with individual concrete storage casks. Six storage concepts were considered in developing the MRS conceptual design, and a complete design was developed for each concept. Each of these designs was evaluated to determine feasibility, cost, relative operational risks, and the time needed for construction. Four of the designs are based on using dry transfer and storage for the spent fuel, one is based on using a water pool for transfer and storage, and one design is based on using a transportable storage cask, which requires no routine transfer of fuel and provides dry storage. The mode of transfer--wet or dry--is important because it has a major bearing on the design of the handling facilities. The six designs were as follows:

1. Dry transfer with storage in vertical concrete casks.
2. Dry transfer with storage in metal casks.
3. Dry transfer with storage in horizontal modules.
4. Dry transfer with storage in transportable storage casks.
5. Dry transfer with storage in vaults.
6. Wet transfer and storage in a water pool.

All six design concepts are considered to be acceptable and feasible. The final choice of concept will be made after a site for the MRS has been found. It will depend on licensing, cost, and schedule considerations and the preferences of the volunteer host.

The MRS must be licensed by the NRC (10 CFR Part 72) and licensing should be expedited by our decision to, in so far as is practicable, incorporate into the MRS design already licensed concepts. We prefer to use technologies already licensed or certified by the NRC or a design that closely approximates that of existing licensed facilities. We plan to select a design that will be suitable for expedited licensing and that will be largely independent of any site-specific conditions.

TRANSPORTATION

We are developing a nationwide transportation system to ship spent fuel and HLW among elements of the waste management system. Waste will be shipped in transportation casks by rail, truck, or barge. Currently, new, higher capacity cask designs are being developed and plans are being made to support shipments of spent fuel to the MRS.

As Federal waste management system receiving facilities become available, OCRWM will need a fleet of shipping casks to carry spent fuel and high level radioactive waste from their present locations to the MRS and final repository. The casks will be rugged containers designed to protect public health and safety by meeting the NRC's transportation regulations (10 CFR Part 71). The NRC transport requirements are similar to those of the International Atomic Energy Agency (IAEA) (Safety Series No. 6).

To obtain these casks, OCRWM has undertaken a major effort in cask development. At present, the primary emphasis is on developing "from-reactor" casks suitable for shipping most of the spent fuel to either an MRS facility or a repository (Initiative 1 casks). Other types of casks that will be needed are those for shipping spent fuel from the MRS to a repository (Initiative 2 casks); casks for special spent fuel that cannot be accommodated in current casks (or those presently under development-Initiative 3 casks); and casks for high-level waste (Initiative 4 casks).

Future Studies

Recent studies have indicated that benefits can be achieved by the use of Multi-Purpose Canisters (MPC) in the waste management system. These canisters would be loaded with the spent fuel at the reactors and sealed. It is intended that once sealed the canisters would be stored, transported, and disposed of without being reopened. The canisters would use separate overpacks for storage, transportation, and disposal. Future work will be directed toward completing designs for canisters, overpacks, and system facilities utilizing Multi-Purpose Canisters.

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