Management of Spent Fuel from Nuclear Power Reactors

Proceedings of an international conference
Vienna, 19–22 June 2006

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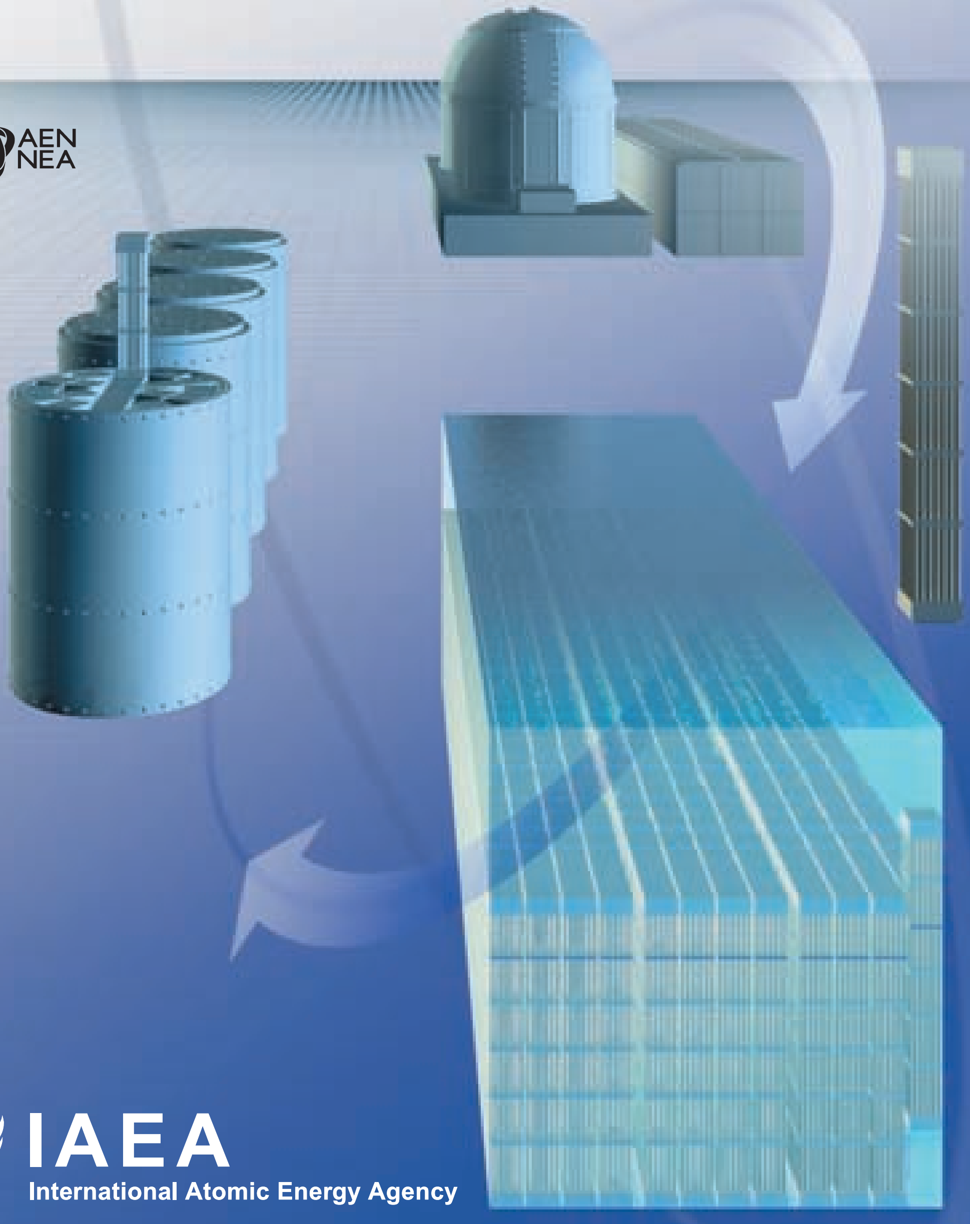

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MANAGEMENT OF SPENT FUEL FROM NUCLEAR POWER REACTORS

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PROCEEDINGS SERIES

MANAGEMENT OF SPENT FUEL FROM NUCLEAR POWER REACTORS

PROCEEDINGS OF AN INTERNATIONAL CONFERENCE
ON MANAGEMENT OF SPENT FUEL FROM
NUCLEAR POWER REACTORS
ORGANIZED BY THE
INTTERNATIONAL ATOMIC ENERGY AGENCY
IN COOPERATION WITH THE
OECD NUCLEAR ENERGY AGENCY
AND HELD IN
VIENNA, 19–22 JUNE 2006

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2007

Sales and Promotion, Publishing Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
http://www.iaea.org/books

Printed by the IAEA in Austria
July 2007
STI/PUB/1295

IAEA Library Cataloguing in Publication Data

International Conference on Management of Spent Fuel from Nuclear Power Reactors (2006 : Vienna, Austria) Management of spent fuel from nuclear power reactors / organized by the International Atomic Energy Agency in cooperation with the OECD Nuclear Energy Agency held in Vienna, Austria, 19–22 June 2006. — Vienna : The Agency, 2007.
p. ; 24 cm. — (Proceedings series, ISSN 0074–1884)
STI/PUB/1295
ISBN 92–0–102107–0
Includes bibliographical references.

1. Radioactive waste disposal — Management — Congresses. 2. Spent reactor fuels — Management — Congresses. I. International Atomic Energy Agency. II. OECD Nuclear Energy Agency. III. Series: Proceedings series (International Atomic Energy Agency).

IAEAL 07–00482

FOREWORD

National strategies for the management of spent fuel vary, ranging from reprocessing to direct disposal. This indicates that spent fuel is regarded differently by countries — as a resource by some and as a waste by others. At the moment most spent fuel is in storage at nuclear power plants, at a few centralized storage sites and at reprocessing facilities. The next steps towards the disposition of spent fuel are either reuse, through reprocessing, or disposal in geological repositories. Because progress on implementing these strategies is slow in most countries, the amounts of spent fuel in storage are increasing. The prospect of a revival of the nuclear power industry in the next decades indicates that even more spent fuel could go into storage. On the other hand, spent fuel has been successfully and safely stored in wet and dry conditions for several decades without serious problems, but without decisions on more permanent solutions there could be the prospect of continued storage for times of up to and beyond one hundred years. The management of spent fuel is, for strategic, economic, safety and security reasons, a key issue for the future of nuclear power and is an issue that many States have yet to decide upon.

The IAEA organized this conference on the management of spent fuel from nuclear power reactors to facilitate the exchange of information on the subject among Member States and to look for common approaches to the issues identified. The conference was organized in cooperation with the OECD Nuclear Energy Agency and held in Vienna from 19 to 22 June 2006. The conference was arranged in eight topical sessions: the evolving international scene, the international safety regime, criticality safety, storage term limits, storage facilities, storage containers, fuel and cladding properties and behaviour, and looking to the future. The conference was structured to promote discussion within the sessions and in focused panel sessions. This publication includes the opening and closing speeches, the keynote papers, the summaries of the panel discussions and sessions, the Conference President's summary and a summary. A CD-ROM containing the unedited contributed papers to the conference can be found at the back of this book.

The IAEA officers responsible for this publication were E. Warnecke of the Division of Radiation, Transport and Waste Safety and W.J. Danker of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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SUMMARY

The IAEA, in cooperation with the OECD Nuclear Energy Agency, held an international conference on the Management of Spent Fuel from Nuclear Power Reactors from 19 to 22 June 2006 in Vienna. The conference was attended by 150 participants and observers from 36 countries and 4 international organizations.

This conference was the latest in a series which started two decades ago. However, at this conference there was an important change of direction — from a technical meeting focused solely on the storage of spent fuel to a strategic meeting on the management of spent fuel, of which storage is only a part.

The sessions of this conference were devoted almost equally to safety and technology aspects. However, before focusing on the specific aspects of technology and safety, a session on the evolving international scene drew attention to the changes of the past few years which have had a strong influence on spent fuel management. A recurrent theme of this session was that there is evidence of a new need and desire for nuclear power in several countries. The reasons for this are various and include the need to secure national energy supplies, to limit the increase of energy costs and to avoid possible global warming by reducing carbon emissions.

If this apparent need for nuclear power is realized in the near future, the new nuclear power plants will be mainly LWRs but with improved safety and economics. The same back end fuel cycle issues as with current power plants can therefore be expected. This implies that the already large amount of spent fuel in storage will continue to increase if no decisions are made on spent fuel management strategies.

Spent fuel is still variously regarded by different countries — as a resource by some and as a waste by others. Consequently the strategies for its management vary, ranging from reprocessing to direct disposal. However, in both cases a final disposition is needed and it is generally agreed that disposal in geological formations is the most appropriate solution.

In all countries, the spent fuel or the high level waste from reprocessing is currently being stored, usually above ground, awaiting the development of geological repositories. While the arrangements for storage have proved to be satisfactory and the facilities have been operated without major problems, it is generally agreed that these arrangements are interim, that is, they do not represent a final and permanent solution. It is becoming increasingly important to have permanent disposal arrangements available so as to be able to demonstrate that nuclear power is sustainable and that it does not lead to an unsolved waste problem. The conference was updated on the good progress being made in several countries towards the development of geological repositories — expected to become available after about 2020.

Recent international fuel cycle initiatives by the USA, the Russian Federation and the IAEA point in similar directions and have similar overall goals:

(a) Improving control over the increasing amounts of spent fuel;
(b) Helping to reduce proliferation and security risks;
(c) Assisting new countries to develop nuclear power.

The initiatives rely on new approaches to processing and recycling, applying advanced technologies to reduce the proliferation risks and to minimize the generation of radioactive waste.

While many larger countries may wish to continue to work on national solutions for nuclear fuel cycle issues, including waste disposal, multilateral solutions may make economic sense to smaller countries. The multilateral approaches also promise better assurances of security and proliferation resistance. The concept of an international safety regime was discussed. It has emerged mainly as a result of the coming into force of the legally binding nuclear related conventions prompted by the Chernobyl accident in 1986. In particular, the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (Joint Convention), together with the International Basic Safety Standards (BSS)[1], may be seen as providing a framework for safety at the international level in the area of spent fuel management. The conference noted that the Joint Convention is an incentive convention and that at the time of the second review meeting of the Joint Convention the contracting parties were not yet prepared to go in the direction of a mandatory approach.

The transport of radioactive material, including spent fuel, represents a particularly good example of the international safety regime. The regulations for transport safety in each country and by each international mode of

[1] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY INTERNATIONAL LABOUR ORGANISATION, NUCLEAR ENERGY OF THE ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

transport are drawn directly from the IAEA Transport Regulations.[2] The safety record in the transport area has been exemplary, as evidenced by the entirely positive results of spent fuel transports over several decades.

There is an obvious linkage between the proposed multilateral fuel cycle initiatives and the international safety regime, that is, any multilateral fuel cycle activities that may be conducted would be expected to comply with the requirements of the Joint Convention and with the recommendations of the IBSS.

It was noted that the BSS in the area of spent fuel management are in the process of being updated and elaborated to cover a wider scope, and during the conference proposals were made on topics that warrant the development of new safety standards, such as on the safety of the long term storage of spent fuel — on methods for its assurance and for the licensing of facilities, and on criticality safety, particularly in relation to the use of burnup credit.

Many technical aspects of spent fuel storage were reviewed during the conference, such as storage facilities, containers, and fuel and its cladding. In relation to these, the topics of burnup credit, long term storage and associated research and development were discussed.

The presentations at the conference pointed out the substantial benefits that can be obtained from burnup credit applications in spent fuel management. However, much of the assessment and development work on this subject has been done in relation to PWR and BWR fuels, and it was clear from the discussions that there is a need for the work to be extended to WWER and RBMK fuels. The conference provided evidence of the trend towards dry storage. While the specialists expressed confidence in the technical development of storage facilities and containers, in order to satisfy long term needs it is clearly necessary for more research and development on fuel behaviour in dry storage. In particular, it was mentioned that high burnup fuels and MOX fuels will need to be carefully assessed in the context of ensuring long term storage safety.

The time period requirements for storage systems have been extended in most countries because of the unavailability of geological disposal facilities. In some countries, new facilities have been built for this purpose; in others, the use of existing facilities is being extended for longer periods. An important safety issue is how to establish the safety of the facilities for long term storage, including consideration of retrieval and transport after storage. There was

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[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material, 2005 Edition, IAEA Safety Standards Series No. TS-R-1, IAEA, Vienna (2005).

much discussion on this subject and it is clearly an area where more research and regulatory work has to be done.

As a conclusion, it seemed to be generally agreed that there should be greater international cooperation on research and development related to the trends indicated during the conference, and that there should be continuing progress towards an international safety regime or, at least, harmonized safety regulations.

OPENING SESSION

OPENING ADDRESS

Y.A. Sokolov
Deputy Director General,
Department of Nuclear Energy,
International Atomic Energy Agency,
Vienna

It is my pleasure to welcome you to Vienna and to this international conference on the Management of Spent Fuel from Nuclear Power Reactors, which is being organized by the IAEA in cooperation with the OECD Nuclear Energy Agency. This conference is a shared effort between the IAEA's Departments of Nuclear Energy and of Nuclear Safety and Security, and is in response to a comment made at the previous spent fuel conference held in 2003 that suggested greater consideration of relevant nuclear safety activities. The conference is an example of one of the key roles of the IAEA, namely to provide a global forum for exchanging information and views on topics important to the peaceful use of nuclear energy.

I also have the pleasure of welcoming you on behalf of our Director General, Mohamed ElBaradei, who I know has a special interest in this week's deliberations. As you are no doubt aware, over the past few years the Director General has persistently and emphatically spoken out in favour of increasing multinational control over proliferation sensitive steps in the fuel cycle, including reprocessing. At this year's General Conference, the IAEA will hold a special event focused on a 'new framework' to facilitate safety, security and proliferation resistance in the future utilization of nuclear energy, including considerations relevant to spent fuel management. One of the precursors of this initiative was the work of the group of experts appointed by the Director General in 2004 to consider options for possible multilateral approaches to the nuclear fuel cycle. Later this morning we will hear from the chairperson of that expert group, B. Pellaud. The Director General sees actions resulting from the expert group's 2005 report as including the establishment of mechanisms to provide assurances with regard to fuel services to countries which are in compliance with their non-proliferation commitments. Such assurances would remove the incentive — and the justification — for each country to develop its own complete fuel cycle. There have been supportive reactions to this initiative. Later today we will hear about the Global Nuclear Energy Partnership of the USA. We will also hear about President Putin's announcement that the Russian Federation is ready to establish international centres that would provide fuel cycle services. Keynote speakers from France and India will follow with perspectives on relevant national developments, including their international implications. After this opening day featuring policies and emerging initiatives in the evolving international scene, tomorrow will focus on nuclear safety. The international safety regime, including results of the recent review meeting of the Joint Convention and the status of the international safety standards, criticality safety and licensing aspects of long term storage of spent fuel are the subjects of these sessions. On Wednesday the focus will shift to spent fuel storage technologies, with sessions sequentially moving from the subject of storage facilities to storage containers and, finally, to the fuel itself. The concluding session on Thursday will draw conclusions and focus on the future. We look forward to making progress this week by clarifying issues, resolving questions, sharing information and insights, and in contributing creative new ideas. We encourage your questions and comments to speakers and poster presenters, as well as your contributions during the panel discussions. Your participation will assist in developing a new vision of future directions for this important topic.

As I mentioned earlier, the scope of this conference has been broadened compared to previous IAEA spent fuel conferences. The conclusions of the conference are intended to be used to analyse, with the help of experts from Member States, the IAEA work programme on spent fuel management and to develop its future directions.

I wish you all a successful and productive conference, and I hope you find some time this week to see the city of Vienna.

OPENING ADDRESS

T. Tanaka
Deputy Director General,
OECD Nuclear Energy Agency,
Paris

It gives me great pleasure to welcome you to this international conference on the Management of Spent Fuel from Nuclear Power Reactors.

This conference series, which the OECD Nuclear Energy Agency (OECD/NEA) has been co-sponsoring for two decades, has now taken an important turn, changing from a technical meeting focused solely on the storage of spent fuel to a strategic meeting on the management of spent fuel, of which storage is only a part.

There is a reason that the conference subject has become more complex than in earlier years. We observe that spent fuel is a very special material. It is a resource to some and it is waste to others. It is one of the most dangerous materials that man produces, yet it is also one that is managed the most safely. Spent fuel also constitutes an important energy reserve and the more so when advanced nuclear fuel cycle scenarios are envisioned. The management of spent fuel thus evokes issues of economics, environmental impact, security, and ultimately sustainable development for our planet. As such, spent fuel is at the core of current concerns and deliberations within the OECD/NEA, whose activities extend from the safety of the nuclear fuel cycle, including disposal, to the development of generation IV reactor systems.

We look forward to being informed of the policy positions being taken in various countries and, in particular, in the USA and the Russian Federation, and to the exchange of experience that will take place in the coming days.

We at the OECD/NEA are glad to have participated in the design and implementation of the conference, in cooperation with the IAEA's Secretariat, whom we commend for their excellent organization of the conference and for the visibility that is given to this important subject. On behalf of the OECD/NEA, I wish all of us a successful meeting.

PRESIDENT'S OPENING ADDRESS

J. Bouchard
Commissariat à l'énergie atomique,
Gif-Sur-Yvette, France

Let me first congratulate the International Atomic Energy Agency and the OECD Nuclear Energy Agency (OECD/NEA) for organizing this conference on the Management of Spent Fuel from Nuclear Power Reactors. It is an honour and a pleasure to act as President of a conference that will address some of the most important and sensitive issues for the safe and sustainable development of nuclear energy.

As Messrs. Sokolov and Tanaka have mentioned in their opening addresses, the scope of the conference is broader than that of the previous spent fuel conferences in this series. However, it is still very important to review the progress made in spent fuel storage technologies, for both containers and the storage facilities themselves. It is also very important to discuss the safety and economic aspects — on the basis of practical experience. It is also the right time to discuss policies and strategies for the long term management of spent fuel. Indeed, behind spent fuel management we have all of the issues related to the back end of the fuel cycle and to radioactive waste management. It is like 'the tree which hides the forest'. Discussing a seemingly innocent topic, spent fuel storage, we open debates on the most political of the choices in nuclear energy policy.

With around 440 nuclear power reactors in operation, among which, more than 360 are LWRs, the world already has some 2 000 000 t of spent fuel in storage, containing approximately 1800 t of plutonium. The fuel is safely stored, either in pools or in dry storage facilities. The facilities are self-protected by their high levels of radioactivity, which limits any risk of diversion and, in fact, from the beginning, particular attention has been given to safeguards measures to prevent spent fuel from becoming a source of proliferation.

The same applies for the countries that are involved in reprocessing and recycling operations. Comprehensive measures are imposed at reprocessing plants or in mixed oxide fuel (MOX) fabrication plants in order to avoid any risk of material diversion and, as a matter of fact, we have not experienced any difficulty in this part of the fuel cycle. The reprocessing and recycling of plutonium in LWRs contributes to a reduction of the amount of spent fuel in storage. It is not the final solution but it helps in the management of most of the waste produced with uranium fuel in LWRs and in the preparations for the future implementation of plutonium recycling in fast reactors.

Thus, the present situation of spent fuel management, either in storage facilities or in the recycling process, is satisfactory, but the total amount of spent fuel is growing every year and the fate of the spent fuel has in most cases not yet been decided.

Furthermore, many countries are considering the possibility of building new power plants. World energy needs are still increasing and will continue to increase in the coming decades. To limit the use of fossil fuels and the dangerous consequences of greenhouse gas emissions, it will be necessary to develop all types of renewable sources, as well as nuclear energy. An increase of the world nuclear capacity by a factor of 3 or 4 during the first half of the century is currently predicted by specialists in energy policy such as the the OECD/NEA or the World Energy Council. Facing the challenge of a substantial increase in the demand for nuclear energy, we see the emergence of Generation III reactors such as the European Pressurised Water Reactors (EPRs) now being built in Finland and France, which rely on the extensive experience obtained with LWRs while bringing new improvements, in particular, to safety. The back end of the fuel cycle for these Generation III reactors should be improved in order to answer the public concerns about waste management and proliferation risks.

Such a growing nuclear energy production capacity will lead to a larger amount of spent fuel. As I mentioned, most of this new capacity will be based on LWRs producing spent fuel very similar to that which comes from present nuclear plants. Thus, if there is no significant change in back end policies, we could be considering a total near to 1 000 000 t of spent fuel in storage by 2050, with a content of close to 10 000 t of plutonium. These approximate figures show that there is a need to have a fresh look at long term options, and not waiting too long before implementing more efficient policies.

Several options have been and are still being considered for the long term management of the spent fuel, ranging from direct disposal to various closed fuel cycle solutions. During the conference, mainly in today's session, we shall have presentations of some national policies. There will certainly be interesting debates about the technical and strategic aspects of these options, including the recent initiatives announced by the USA, the Russian Federation and the IAEA.

Technical issues associated with the back end of the fuel cycle are also an important part of the international cooperation for the future development of nuclear energy. Two forums are working towards this goal. The first one is INPRO, the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles, with 40 countries considering the user requirements, the safety and security aspects of innovative systems and the merits of various proposals for such systems. The second one is GIF, the Generation IV International Forum, which started as an initiative of the USA in 2000 and which now has 11 members working on cooperative research and development programmes on future nuclear systems. It is interesting to note that both forums are focusing on a closed fuel cycle with full recycling of all the actinides in fast neutron reactors.

At the present time, spent fuel management and the back end of the fuel cycle are probably the most important issues for the future of nuclear energy. We have safe reactors producing electricity competitively. The industrial developments in the coming decades, with Generation III reactors, will bring still more improvements in safety and competitiveness. There are also many projects involving the development of innovative reactors that could, if successful, help to deal with new demand. Some of them address other possible applications of nuclear energy than electricity production, such as the large scale production of hydrogen or the industrial use of heat at high temperatures. Others are aiming at facilitating the implementation of nuclear energy in developing countries, which have different kinds of infrastructures from those in industrialized countries.

However, all of these reactors will still produce spent fuel containing not only fission products but also actinides and, in particular, the most important element for long term activities, plutonium. Recycling this element in thermal reactors, for example, through MOX fuel, offers some important advantages but does not bring a final answer because, although it reduces the amount of spent fuel, the spent fuel contains a higher concentration of plutonium.

The sensitivity of the waste management issue and its effect on public opinion in many countries has been at the origin of important research and development programmes on partitioning and transmutation. The burning of plutonium and other actinides is considered today to be an important strategy for achieving sustainability — together with the need to make better use of natural resources.

Whatever the choice made by each country for the future management of spent fuel, safety and proliferation resistance will remain of paramount importance for the current use of nuclear energy and for its further development. The conference will address many aspects of these issues. Nevertheless, I would like to emphasize the following remark. When looking to new developments we tend first to solve technical issues in order to reach adequate performances and then to find solutions for satisfying the requirements of economics. Safety and non-proliferation issues have often been considered in the next step after these first two. Nowadays, we have become convinced of the need to consider these issues from the beginning of the development process and this is the way we are proceeding, for example, in the GIF programme.

Let me conclude these few introductory remarks by wishing you a very fruitful conference. The programme committee and the scientific secretaries have done a good job of selecting presentations which, I am sure, will be of high quality and lead to lively discussions on topics of great importance for the present and the future of nuclear energy.

SPENT FUEL MANAGEMENT — THE EVOLVING INTERNATIONAL SCENE

(Session 1.A)

Chairperson

J. BOUCHARD
France

INTERNATIONAL PERSPECTIVES ON SPENT FUEL MANAGEMENT

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Abstract

The paper presents an overview of the international scene in the area of spent fuel management, describing the general global situation, achievements at the national level, ongoing initiatives towards multilateral solutions for the back end of the fuel cycle, and the status and developments in relation to achieving an international regime for assuring safety in this area. Finally, the activities of the International Atomic Energy Agency in the context of its work in the areas of technology and safety are described.

1. INTRODUCTION

The production of nuclear electricity results in the generation of spent fuel that requires safe, secure and efficient management. Appropriate management of the resulting spent fuel is a key issue for the steady and sustainable growth of nuclear energy. At the end of 2005, 443 nuclear power reactors were operating in 30 countries worldwide [1], providing 16% of the global electricity supply. Over 10 000 t of heavy metal (t HM) are unloaded from these reactors each year, which will increase to ~11 500 t HM by 2010. This is the largest continuous source of civilian radioactive material being generated, and needs to be managed appropriately.

Originally all spent fuel was expected to be reprocessed within a few years and the remaining fuel material recycled into new fuel. The waste from reprocessing was intended to be disposed of in geological repositories. Policies have, however, changed over the years. Some countries are continuing the recycling route, while others have decided to regard the spent fuel as a waste intended for direct disposal. Most countries have adopted a wait and see position. With this situation, spent fuel storage for extended durations is becoming a reality. Member States have referred to storage periods of 100 years and even beyond, and as storage quantities and durations are extended, new challenges arise in the institutional as well as in the technical area. Recently several new initiatives have been taken to increase international cooperation in the study and implementation of the recycling of spent fuel in a way that takes due regard of safety, security and non-proliferation of sensitive nuclear technology.

2. SPENT FUEL ARISINGS

The total amount of spent fuel generated worldwide in the 52 year history of civilian nuclear power is around 280 000 t HM, of which roughly one third has been reprocessed, leaving around 190 000 t HM of spent fuel, mostly in wet storage pools but with an increasing amount in dry storage. Figure 1 shows how the amounts of spent fuel generated, reprocessed and stored around the world have evolved since 1990 and includes projections to 2020 [2]. The total amount of spent fuel that will be generated by 2020 is estimated to be 445 000 t HM. Regional projections reported by the IAEA [3] are shown in Fig. 2.

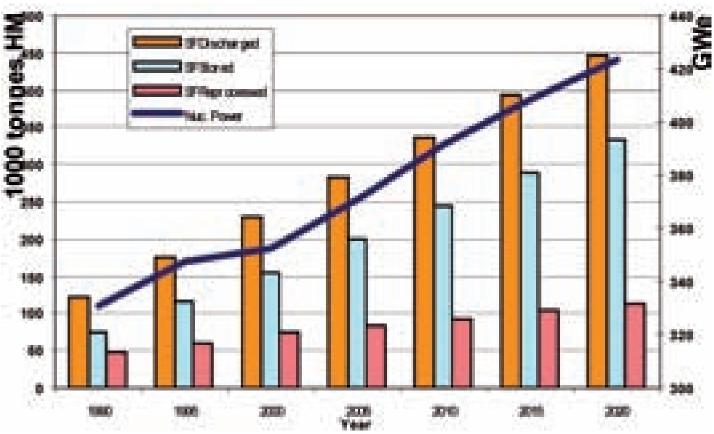

FIG. 1. Cumulative spent fuel discharged, stored and reprocessed from 1990 to 2020.

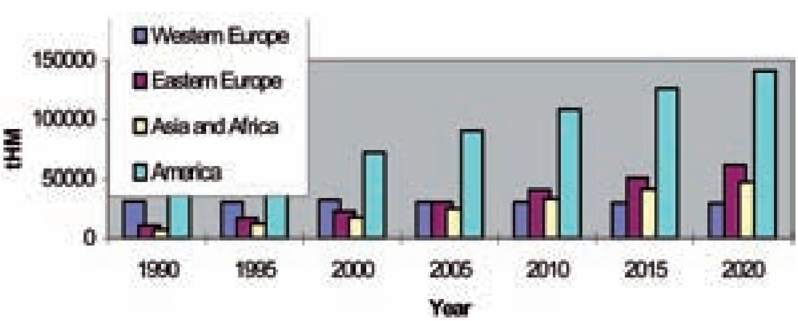

FIG. 2. Spent fuel stored by regions.

3. REPROCESSING OF SPENT FUEL AND RECYCLING

Some Member States have adopted a policy of reprocessing spent nuclear fuel and subsequently recycling the uranium and plutonium as mixed oxide (MOX) fuel in LWRs. Commercial reprocessing and MOX fuel fabrication capacities only exist in a few Member States. Table 1 shows the operating commercial size reprocessing facilities in the world. Other countries, e.g. Belgium, Germany and Switzerland, are involved in recycling material from their own reactors that have been reprocessed in France or in the United Kingdom.

TABLE 1. COMMERCIAL SIZE REPROCESSING PLANTS

Name	Country	Capacity (t HM/a)	Type of fuel
BNFL Magnox	UK	1500	Magnox
BNFL Thorp	UK	900	LWR, AGR
JNC Tokai	Japan	210	LWR
La Hague — UP2-800	France	1000	LWR
La Hague — UP-3	France	1000	LWR
RT-1 Mayak	Russian Federation	400	LWR
Rokkasho [a]	Japan	800	LWR

[a] The Rokkasho plant is being commissioned, with a planned start of commercial operation in 2007.

4. THE CHANGING ENVIRONMENT FOR SPENT FUEL MANAGEMENT

Two separate trends can be seen in spent fuel management. On the one hand, the trend towards more storage capacity for longer durations is expected to continue. At the same time, the interest in recycling in a longer time perspective is increasing around the world. The latter trend is connected to the increased activities aimed at improved utilization of the uranium resource through recycling in fast reactors.

A third trend that needs to be considered in this context is the increasing expectation that use of nuclear energy will expand in countries that already have nuclear energy, as well as in some new countries. This would highlight the need for new approaches and solutions to spent fuel management and, in particular, the need for increased cooperation between countries both regionally and worldwide.

The trend towards more long term storage is complicated by trends towards higher initial enrichment and higher fuel burnup, as well as other considerations including the use of new fuel designs and MOX. Given the importance of effective spent fuel management for the sustainable utilization of nuclear energy, Member States of the IAEA are maintaining an active interest in related work, as evidenced by high participation in IAEA sponsored meetings on the subject.

Given this situation, increased storage capacities are being created around the world, in the form of both 'at reactor' and 'away from reactor' facilities. Wet fuel storage in water pools, which has been used for more than 50 years, is a mature technology that will continue to play a major role in spent fuel storage. At the same time, dry spent fuel storage technologies are being used more and more, especially in relation to longer term storage. Dry storage facilities employ a variety of configurations, including modular vaults, silos and casks. Figure 3 shows the casks at the ZWILAG facility in Switzerland and the Fort St. Vrain vault in the USA.

The extended storage time for the spent fuel raises a number of issues concerning long term safety and security, such as the long term behaviour of

(a)

(b)

FIG. 3. Dry fuel storage technologies: casks at the Fort St. Vrain vault in the USA (a) and at the ZWILAG facility in Switzerland (b).

fuel and of the components of the storage facilities, as well as issues connected to the long term management of information. While no significant safety problems are expected, it is important to monitor the facilities, to learn from experience with them and to apply the results in designing and operating new facilities, from the beginning, for extended storage. As the storage facilities will contain substantial amounts of radioactive material, concerns about nuclear security have also to be carefully considered.

The long term storage of spent fuel provides a suitable buffer in the spent fuel management system and also provides time for the consideration of what should be the next step — disposal or recycling. Such a decision will depend on national policies and on technical developments worldwide.

No geological repository for spent fuel or high level waste (HLW) has yet been built. The only operating geological repository is the Waste Isolation Pilot Plant (WIPP) in the USA, which is used for disposal of low level transuranic (long lived) radioactive waste. In addition to WIPP, good progress has been made in several countries, notably Finland, France, Sweden and the USA, on repositories for HLW or spent fuel from commercial nuclear power plants. However, none is expected to start operation until around 2020. The Finnish, Swedish and US repositories are intended for spent fuel.

During the past few years there have been increased research and development activities on the recycling of spent fuel. With the aim of facilitating the long term sustainability of nuclear power there have been increased activities on advanced reactors and on advanced fuel cycles. Several initiatives for multilateral cooperation were taken a few years ago, e.g. the IAEA coordinated international project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) [4] and the Generation IV International Forum (GIF) [5]. Both INPRO and GIF include recycling in advanced fast reactors as important components. More recently, in 2006, 2 important initiatives were presented by the Russian Federation and the USA, both of which have the potential of changing spent fuel management strategies. These are the Russian proposal for an international fuel cycle centre [6], under IAEA control, and the US Global Nuclear Energy Partnership (GNEP) [7]. Both include an important component of non-proliferation of sensitive nuclear technology at the same time as an increased emphasis on recycling. The recycling would provide a more efficient use of uranium resources and also have the potential to reduce the volume requirements for geological disposal, as the heat generation of the waste can be reduced. More information on these new initiatives will be given in other presentations at this conference.

The proposed multilateral approaches raise a number of technical, legal and political issues. In September 2003, IAEA Director General M. ElBaradei proposed to take a fresh look at multilateral approaches. He set up an independent international expert group on Multilateral Approaches to the Nuclear Fuel Cycle (MNA) under the chair of B. Pellaud. The results of the expert group's discussions will be presented in another paper at this conference [8].

The expert group, the Russian President's initiative and the US GNEP activity all point in a similar direction and have similar overall goals:

(a) Obtaining control over the increasing amounts of spent fuel;
(b) Makins a contribution to the reduction of proliferation risks;
(c) Assisting new countries wishing to develop nuclear power.

The IAEA welcomes such initiatives and is willing to contribute actively to such approaches in cooperation with, and upon the request of, Member States. More details on international/multilateral approaches will be presented in other papers in this conference. One important point is that the long term and continued national commitment to these approaches by the countries involved must be maintained to ensure credibility. In addition to political stability, other factors have to be taken into account, in particular the availability of the necessary facilities and technologies, including those for the disposal of radioactive waste. The national public acceptance of international approaches must also be considered. This may become a critical issue and it is one that has not yet been fully addressed.

The number of spent fuel transports over large distances may be expected to increase as more and more spent fuel is transported from reactors, to stores and finally to disposal sites. Although from the viewpoint of experts in the field the safety record of transports is very convincing, public perception, at least in some countries, is different. The question of public acceptance of the transport of spent fuel over large distances cannot be overlooked and has to be taken into consideration from the beginning.

5. INTERNATIONAL CONVENTIONS AND SAFETY STANDARDS

5.1. Joint Convention

The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management came into force in 2001. It represents a commitment by the participating States to achieve and maintain a consistently high level of safety in the management of spent fuel and of radioactive waste [9]. At present, 41 States have ratified the Joint Convention.

Although this represents less than 30% of the IAEA Member States, these countries contain about 97% of the commercial spent fuel being generated.

Details of the results of the second review meeting of the Joint Convention will be presented in another paper [10]. During the review meeting the Contracting Parties demonstrated their commitment to improving policies and practices, inter alia, their national strategies for spent fuel management and their engagement with stakeholders and the public. Indeed, it was stated in the report of the review meeting that "many Contracting Parties have already developed, or are currently developing, spent fuel and waste management strategies based on increasingly comprehensive inventories".

The success of the Joint Convention will depend on the rigour with which the reviews are performed in order to make accidents and incidents in the area of spent fuel and radioactive waste management as unlikely as possible. Learning from experience and improving the review process with time is essential. It is also important that all countries managing spent fuel and radioactive waste ratify the Joint Convention in order to benefit from the experience of other countries.

5.2. Safety standards

Safety standards are the foundation that underpins the international safety regime and, in that regard, the development of a complete corpus of safety standards is one of the key factors for assuring that safety is maintained in the spent fuel management area.

Currently, most spent fuel is stored on-site at nuclear power plants. In addition, spent fuel in some countries is stored at reprocessing facilities or other centralized locations. Both of these storage strategies can be expected to be in place for the next several decades, and the safety standards have to address both situations.

Although there are numerous safety standards concerned with different aspects of safety at nuclear power plants, for research reactors and radioactive waste management, there are only a few which address spent fuel management. The existing safety standards on spent fuel storage are more than 10 years old and need to be updated. Also, no comprehensive guidance on criticality safety is available within the safety standards series. It is necessary, therefore, to take a fresh look at the needs, at the international level, for safety standards and guidance for spent fuel management.

The transport of spent fuel has been a practice for many years and the safety standards for the transport casks are prescribed in the IAEA Regulations for the Safe Transport of Radioactive Material [11]. The developments taking place in the combined use of spent fuel casks for storage and transport and possibly even for disposal will require a new approach. The design standards for long term storage casks and those for transport casks need to be harmonized.

6. EMERGENCY PREPAREDNESS

In addition to working to ensure that spent fuel management activities are carried out in as safe a manner as possible, the concept of defence in depth demands that appropriate emergency preparedness activities be implemented to cover the possibility of a failure in the operational safety system. Emergency preparedness is the final barrier to ensuring the safety of workers and the public. It includes actions to prevent, to mitigate and to respond to unanticipated transients, which means, in the context of spent fuel management, being able to address the potential emergency situations associated with fission control, heat removal and radiation safety.

For all practical purposes, the emergency preparedness arrangements associated with the operation of a nuclear power plant or a research reactor fully encompass the spent fuel issue. They include being able to address unplanned events in spent fuel pools, in dry cask storage facilities, at reprocessing plants and during transportation. All nuclear installations are designed to include a strategy on how to safely store spent fuel on-site until it can be moved to a more permanent location. Additionally, since the events of 11 September 2001 in the USA, the vulnerabilities of spent fuel storage facilities to external threats have been reconsidered. Many compensating measures have been put into place in this regard — this includes improvements in security measures (as a preventative action) and improvements in the safety measures to preclude any incidents or events (reactive actions).

It is also important to have response measures that are appropriate and well exercised. Such measures must address both the security and safety considerations associated with spent fuel. Integrated response plans have been developed and rehearsed, and improvements in the design and maintenance of backup systems have been undertaken.

Given the long time perspective for storage, it is necessary to provide assurance that the final barrier, the emergency plan implementing procedures and the bases for the plan are not forgotten. The commitment to ensuring a viable emergency preparedness and response programme during the possibly prolonged periods of time associated with storage must be maintained.

The expected increased number of transports of spent fuel, sometimes over long distances, raises some new issues, possibly involving States that do not operate nuclear power plants and emergency response centres. Along the routes of transport, a sufficient number of emergency response centres that could reach the site of the emergency at short notice to provide 'first aid' should be established. These response centres should be linked to a central emergency response agency which has the capability to organize the required personnel and equipment for the affected site within a short period of time.

7. IAEA ACTIVITIES IN SPENT FUEL MANAGEMENT

The IAEA has started to revise and expand its safety standards in the area of spent fuel management. Recently, the IAEA's Commission on Safety Standards gave its approval for the development of three IAEA Safety Standards related to spent fuel management. The tentative titles are as follows:

Safety Requirements:	Safety of Fuel Cycle Facilities;
Safety Guides:	(a) Storage of Spent Fuel;
	(b) Safety of Reprocessing Facilities.

However, consideration has to be given to the completeness of the standards in this area and to whether all key safety related aspects are being addressed. For example, it needs to be considered whether the standards properly address issues related to: ensuring the long term safety of stored fuel and the associated process for licensing, especially when new and high burnup fuels are to be stored, to ensuring safety when storage capacities are increased by various means, including the use of burnup credit, and to ensuring safety in the transport of spent fuel, taking due account of the need to move spent fuel after long storage times, of high burnup fuel and of possibly new fuel designs. The findings of this conference are expected to be a useful input to the future development of standards in this area.

In the application of safety standards, the activities of the IAEA are focused on how best to apply the safety standards that have been and are being developed. Currently, this is being pursued through the IAEA Safety Review Services, specifically in the areas of operational and design safety. The operational safety services for power reactors, research reactors and fuel cycle facilities have modules that include the review of the safety of the spent fuel management programmes. The operational safety services for power and research reactors have been in place for several decades, while the fuel cycle facility programme will be inaugurated in 2007. Lessons learned and operational best practices are captured in the review reports and databases related to these programmes. Improvement of the operational safety services and the information exchange arrangements will remain key IAEA initiatives for the foreseeable future.

The IAEA has been proactively involved in the technological aspects of spent fuel management activities for many years. Various meetings have been organized, often focused on producing technical documentation available to all Member States on topics of interest [12]. Most IAEA technical publications can be accessed and downloaded from web site http://www-pub.iaea.org/MTCD/publications/tecdocs.asp.

Spent fuel storage technology (particularly dry storage) is undergoing an evolution, with modified and new fuels, new designs and increasing target burnup levels. Increased burnup infers higher strains and increased cladding hydriding and oxidation. The Coordinated Research Project on Spent Fuel Performance Assessment and Research (SPAR) addressed the research needed to justify spent fuel storage for very long periods of time (more than 50 years). Building on three earlier BEFAST projects (behaviour of spent fuel and storage components during long term storage), SPAR efforts began in 1997 and resulted in a technical report published in 2003 [13]. The SPAR studies are continuing until 2008 with topics that include surveillance and monitoring programmes for spent fuel storage facilities, fuel materials performance evaluation for wet/dry storage, and the collection and exchange of spent fuel storage experience.

A technical meeting organized by the IAEA in Ljubljana, Slovenia, in October 2004 explored provisions for long term storage of spent power reactor fuel by focusing on the evolution of national approaches, operational considerations and cooperative initiatives. Based on a recommendation from the Ljubljana meeting, the IAEA initiated an activity to investigate the handling of damaged spent fuel, with a technical meeting held in December 2005. The chairperson of the meeting in his summary emphasized that damage is not an intrinsic property of the fuel. Depending on its functional requirements, fuel considered damaged for one phase may not be considered damaged in a subsequent phase.

With increasing initial enrichments and longer storage times, different approaches to increase the effectiveness of the storage capacity have become more important. One such approach is burnup credit. Criticality safety analyses of spent fuel systems traditionally assumed that the fuel was fresh, resulting in significant conservatism. Improved methods (calculations and measurements) for developing knowledge of spent fuel characteristics support efforts to take credit for the reactivity reduction associated with fuel burnup by reducing the conservatism while maintaining appropriate criticality safety margins. The IAEA started work in this area in 1997 and since then several technical meetings have been held. A report [14] published in 2003 gives details of the discussions on the progress and status of burnup credit applications for spent nuclear fuel.

As storage durations increase, attention to maintenance is crucial. The accumulated industrial experience of the past several decades in the operation and maintenance of spent fuel storage and transport casks and containers was summarized in a meeting involving representatives of operators, regulators and other stakeholders.

Another report concerns the optimization of cask/container loading for long term spent fuel storage and is based on several meetings that obtained views from both regulators and implementers.

Effective management and protection of storage related data is a key condition for long term spent fuel management. As data storage technologies evolve and as personnel rotate, maintaining the continuity of knowledge will require continuing attention. A publication on data requirements and maintenance of records for spent fuel management was to have been published in 2006.

A particular challenge facing countries with small nuclear programmes is to prepare for extended interim storage and then disposal of their spent nuclear fuel. Accordingly, the IAEA organized meetings on technical, economic and institutional aspects of regional spent fuel storage. The conclusions were issued in 2005 [15]. The main conclusion is that technical considerations and economic issues may be less significant than ethical and institutional issues for the development of a multinational project. Based on prior meetings, a report providing guidance on methodology and selection criteria for away from reactor storage facilities was to have been issued in 2006.

Interest has been growing recently in emerging technologies for spent fuel treatment. In response to this, the IAEA held a meeting in October 2005 to review spent fuel treatment options and applications.

8. IMPORTANT ISSUES AND FUTURE DIRECTIONS

The different approaches to spent fuel management that will be discussed during this conference are expected to raise a number of issues of a technical, regulatory, institutional and legal character on which the IAEA would appreciate guidance, e.g.:

(a) Technical, safety and security implications of more effective storage concepts, e.g. high density racking or dry cask storage, and implications of long storage times;

(b) Safety, security and non-proliferation implications of increased recycling activities, including a substantial increase in transports;
(c) Public acceptance aspects of increased international trade and transport;
(d) The needs for international standards for the multilateral nuclear fuel cycle facilities and strategies, while keeping national independence;
(e) Legal and regulatory frameworks for multilateral approaches, including international oversight;
(f) Need for international exchange of experience and the role of the IAEA.

The two separate trends that were identified initially, longer term storage and increased interest in recycling, will require active cooperation and involvement between organizations in Member States at bilateral, multilateral and international levels. They highlight the need for the exchange of technical information and guidance, and the need to take a fresh look to ensure that the relevant safety standards will be in place as new options and facilities develop. The results of this conference will help to guide the IAEA in its planning for future activities in the area of spent fuel management and recycling.

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DISCUSSION

A. GONZÁLEZ (Argentina): About 20 years ago, the IAEA was involved in an exercise called the International Nuclear Fuel Cycle Evaluation (INFCE), one of whose conclusions was that there is a big difference in radiological impact between the open and the closed nuclear fuel cycles, the latter having the smaller impact. Politicians have come and gone during the past 20 years, but the fact that this difference exists has not changed. However, I missed that point in Mr. Forsström's presentation. I mention that because, in my view, the IAEA has technical obligations independent of political considerations.

H.G. FORSSTRÖM (IAEA): I agree that the difference still exists. However, whichever nuclear fuel cycle option one chooses, one has to ensure safety, and the optimum from that point of view may well differ from country to country.

T. TANIGUCHI (IAEA): In my opinion there are two major challenges in connection with the Joint Convention — bringing further States, particularly ones with nuclear power plants, into membership and increasing the usefulness of the Joint Convention process as a tool for improving safety.

AN INTERNATIONAL PERSPECTIVE ON THE MANAGEMENT OF SPENT FUEL

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Abstract

The paper addresses some of the political, strategic and technical issues associated with spent nuclear fuel and its management. In particular, it discusses the issues surrounding the long term storage of spent nuclear fuel and reviews the ongoing work to address these issues. Finally, it summarizes the programmes of work at the OECD Nuclear Energy Agency aimed at addressing this and related subjects.

1. INTRODUCTION

Among the important challenges within the field of nuclear energy is that associated with the management of radioactive waste from nuclear power plants, and especially the management of spent nuclear fuel. In this context it may be useful to review the fundamental aspects of the issues and challenges and to discuss them in a broader framework than that in which they are normally considered.

2. WHY ARE WE CONCERNED ABOUT SPENT NUCLEAR FUEL?

From some perspectives, spent nuclear fuel might seem to be a small, even inconsequential issue. The total worldwide volumes of spent fuel, for example, are relatively small. The volume of radioactive waste is dwarfed by the volume of total industrial toxic waste, and spent nuclear fuel, in turn, comprises a small percentage of the total radioactive waste volume. In the UK, for example, spent nuclear fuel comprises only 2% of all radioactive waste by volume from nuclear power plants. According to figures from the US government, after more than 30 years of operation the nuclear power plants in the USA (more than 100) have produced a total volume of just over 50 000 tons of spent nuclear fuel, which is quite small compared with the over 98 million tonnes of waste ash produced annually from electric utilities fuelled by coal.

The cost of management of spent nuclear fuel also represents a small fraction of the total cost of nuclear power generation: According to a recent OECD/NEA study, the cost of spent nuclear fuel management accounts for only 1–5% of the cost of nuclear power generation. From a cost perspective then the power industry is relatively insensitive to strategies and policies concerned with spent nuclear fuel management.

Yet spent nuclear fuel has characteristics and hazards that make its management particularly challenging. Although it constitutes only a small fraction of the total radioactive waste volume in national waste inventories, it contains most of the radioactivity. In the UK about half of the radioactivity in the national radioactive waste inventory can be attributed directly to spent nuclear fuel — and this rises to over 97% if high level radioactive waste and plutonium, as by-products of spent nuclear fuel reprocessing, are included. The combination of the fission products and the actinides associated with spent nuclear fuel require that attention be paid to near term safety issues associated with potential external radiation exposure and to heat generation, as well as to the management and containment of extremely long lived radionuclides.

Furthermore, the management and disposal of spent nuclear fuel is at the heart of the debate on the future of nuclear power. It seems likely that there will be a worldwide resurgence in nuclear power generation in response to national desires for greater energy security and independence. The availability of petroleum may depend on political regimes that are unstable or hostile, and its price and supply may therefore fluctuate. In contrast, the most recent OECD/NEA–IAEA 'Red Book' shows that uranium resources are distributed globally and that they can provide fuel supplies for an estimated 250 years for all existing nuclear power plants. This could be readily extended to thousands of years by adopting reprocessing and advanced fuel cycles. Concerns over global warming are also prompting a re-evaluation of nuclear power.

However, the failure to achieve meaningful progress in establishing and implementing disposal solutions for radioactive waste and spent fuel is a key issue in the argument against the expansion of nuclear power. This contributes to perceived risks and a reluctance (in market driven economies) to invest in nuclear power. There is concern about the potential security and non-proliferation risks associated with spent nuclear fuel storage. In the USA, for example, considerable attention has been devoted to evaluating the safety of spent nuclear fuel in storage pools if threatened with terrorist attacks. The end result of these various competing concerns is that the debate continues worldwide about strategies for the management, processing, storage and disposal of spent nuclear fuel.

3. WHERE DO WE AGREE REGARDING MANAGEMENT OF SPENT NUCLEAR FUEL?

It is important to acknowledge that storage of spent nuclear fuel is already being successfully implemented today. With active surveillance and maintenance, the safety and security of storage can be relied upon for the near term and for decades into the future.

When considering the broader perspective it should also be remembered that there is, in fact, already wide agreement on the basic principles of the management of spent nuclear fuel. These principles are formalized in the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (the Joint Convention).

There is also affirmation that geological disposal is the only sustainable strategy available today that provides long term protection from the hazards of radioactive waste. The OECD/NEA's Collective Opinions (in 1991 and 1995) concluded that geological disposal:

(a) Is responsive to fundamental inter- and intra-generational equity concerns;
(b) Is technologically feasible and sound, and that assessment methods are available today to evaluate adequately and give confidence in safety, now and in the future;
(c) Represents a 'sustainable' solution because it is both passive and permanent;
(d) May be implemented in a step-wise manner to leave open the possibility of adaptation to societal progress and demands.

These conclusions remain valid today. All countries that have made a policy decision on the final step for the management of spent nuclear fuel have selected geological disposal as the end point. Furthermore, in the light of increased concern regarding terrorist incidents and other threats, geological disposal offers protection from human intrusion. An IAEA position paper on the topic stresses that putting hazardous materials underground increases their security (The Long Term Storage of Radioactive Waste: Safety and Sustainability (2003)).

There are, however, significant changes emerging in the respective roles of storage and disposal. These and other factors indicate that there is a need to re-examine the strategies and priorities for management of spent nuclear fuel.

4. WHAT IS AN ACCEPTABLE END POINT FOR THE MANAGEMENT OF SPENT NUCLEAR FUEL?

Most spent fuel is currently being stored at nuclear power plant sites, while the remainder is being stored at a few centralized storage and reprocessing facilities. While there is endorsement of the concept of geological disposal in national policies, no country has yet succeeded in operating a deep geological repository for spent nuclear fuel. Some countries have recently undertaken a reassessment of their national positions regarding waste disposal and the UK, Canada and France have engaged in extensive programmes of public debate, information and evaluation to define and examine options for radioactive waste management — and for spent nuclear fuel and high level waste in particular.

From these developments it can be seen that the line is being blurred between the purposes of storage and disposal. Traditionally, storage has been used for a variety of purposes, including:

(a) Decay storage — to allow levels of radioactivity and heat output to decline.
(b) Buffer storage — to provide stock for an ongoing process, such as transport or immediate disposal.
(c) Interim storage — to await the required facility or transport capability becoming available.
(d) Strategic storage — for materials that have some potential future use or value.

Historically, storage for these various purposes has been envisioned and designed for periods of time ranging from days or weeks to tens of years. Recently, however, there has been consideration of storage for periods of the order of 100 years or more. The Canadian national programme recently recommended, for example, the implementation of 'adaptive phased management', an approach that centralizes waste emplacement in a facility deep underground but specifies no end point for the retrieval phase. The implementing agency states that "A future society will decide whether and when there is sufficient confidence in the safety of the approach to seal and backfill the repository".

These approaches are motivated by various factors. Foremost among these are: (i) the need to gain greater public acceptance for disposal strategies and (ii) to allow for the possible development of disposal and treatment technologies. For spent nuclear fuel in particular, development of advanced reactor designs and fuel cycles may also be relevant.

Faced with these debates and developments, the question must be asked, "Is storage an acceptable end point for the management of spent nuclear fuel?" The answer is that storage itself cannot be an acceptable end point. The International Conference on the Safety of Radioactive Waste Management held in Cordoba, Spain, in March 2000, concluded as much, stating that: "perpetual storage of radioactive waste is not a sustainable practice and offers no solution for the future". Further, the use of storage as an end point contradicts the definition of storage as used in the Joint Convention.

Nevertheless, it is necessary to accept that expanded roles for storage are being considered and to think concretely about how to address this strategy for management.

5. WHAT ARE THE PARTICULAR CHALLENGES WITH VERY LONG TERM STORAGE?

The extension of storage schemes and designs beyond decades to a century or more implies the need for many changes in thinking and technical approach. From a technical perspective, attention must be devoted to properly maintaining facilities, and to the possible need for their refurbishment (especially those facilities that may be extended beyond their original design lives) or to the repackaging of waste. Consideration has also to be given to the potential security risks associated with long term storage.

It is important to recognize that the challenges for spent fuel management lie not only in the technical aspects, but also in the societal aspects. Consideration has to be given to the stability of future societies and to the need for infrastructure and resources to support storage facilities in the distant future. As the design life of facilities is extended, it becomes progressively more difficult to estimate the associated costs.

From a regulatory perspective, consideration will have to be given to structuring the licensing process, for example, the terms for review and renewal of licenses and the provision of oversight over long time frames. Furthermore, there will be uncertainties about the stability of relevant organizations and agencies over these time periods. This raises questions about who has (or should have) ownership of the spent fuel over longer time frames. It also affects the assessment of future liabilities, which can be a potentially significant issue affecting investments (e.g. for proposed new power plants) in market economies.

The societal challenges are not to be underestimated. An inability to guarantee adequate funding or institutional stability and control in the face of these uncertainties has a direct impact on the confidence to ensure safety in the future.

6. WHAT IS BEING DONE TO ADDRESS THESE CHALLENGES?

These challenges are being addressed in national programmes. First and foremost, careful oversight is being maintained to ensure the safety and security of existing stores of spent nuclear fuel. Important work is also being done within national programmes to evaluate storage concepts (i.e. wet storage vs. dry storage), to assess and mitigate any security risks, and to refine design concepts to account for the effects of extended periods of storage and of retrievability, and to facilitate the transition from storage to disposal.

Secondly, progress is continuing towards defining and implementing ultimate disposal solutions for spent nuclear fuel. There are open and transparent discussions in many countries directed towards building public understanding and acceptance of national policies. Other countries have established national policies and are moving ahead with institutional planning, site evaluations, research programmes, and preparations for licensing procedures.

Thirdly, in several national programmes investments are being made in the development of new technologies focused on reducing the volume or hazard of spent nuclear fuel and related waste, as well as in providing enhanced security and allowing for the recovery of the residual energy potential in spent nuclear fuel. The most recent example of this is the Global Nuclear Energy Partnership (GNEP) initiative being pursued in the USA.

As a reflection of the priorities of its member countries, the OECD/NEA Nuclear Energy Agency is working actively in these same areas. International collaboration is aimed at defining the issues, sharing successful experience and providing guidance. Key activities and initiatives of the NEA address:

(a) The evolving roles of storage.
(b) The definition and demonstration of the safety of disposal:
  — Current initiatives explore and define the bases for national approaches to long term safety criteria and the treatment of different timescales, of the order of thousands to millions of years.
  — Activities to establish and strengthen the technical and scientific bases for safety cases, including the definition of FEPs (features, events and processes), the integration of geological information, and the optimization cycle for engineered barrier systems. An international symposium (co-sponsored by the IAEA and the OECD/NEA) in January 2007 will

take stock of recent advances and overall progress in safety cases for geological disposal.

— The OECD/NEA also provides peer reviews of national programmes at key stages in their progress towards implementing disposal solutions, the most recent example being a review of the French 'Dossier 2005 Argile' for a reversible disposal concept in a clay formation.

(c) Methods for building societal confidence, including tools and methods for defining stakeholder interests and needs, undertaking constructive dialogue, establishing trust in regulatory agencies and building public acceptance of decisions. While much of the OECD/NEA's work has focused on geological disposal, the lessons learned regarding societal dialogue and acceptance are readily transferable to related areas, including selection of radioactive waste management strategies, siting decisions, and regulation and oversight of all types of nuclear facility.

(d) The scientific bases underlying spent fuel generation and management, including the safety of the nuclear fuel cycle and participation in international cooperative projects such as the Halden reactor.

(e) Assessment and support for the development of key technologies such as advanced fuel cycles and partitioning and transmutation. A recent study, for example, specifically examined the implications for waste management of various potential advanced fuel cycles. The OECD/NEA also serves as a facilitator and secretariat to the GENERATION IV initiative.

This conference is another example of the excellent collaborative work being done internationally on spent nuclear fuel management. These efforts are important for ensuring that progress is made in establishing and implementing ultimate disposal solutions for spent nuclear fuel and for maintaining and improving the current successful management and storage techniques to ensure they stay safe and secure.

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DISCUSSION

J. WHANG (Republic of Korea): Has the OECD/NEA looked into the question of the possible very long term storage of spent fuel?

T. TANAKA (OECD/NEA): Yes, and a paper on the role of storage in the management of long lived solid radioactive waste and spent fuel is going to be published by us soon. Pre-prints are already available.

J. WHANG (Republic of Korea): Our experience in seeking possible sites for away-from-reactor spent fuel storage facilities has been that the people in the local communities which we have visited want to know how long the facility will be designed for.

That has been the experience also of colleagues in Japan, where people in the city of Mutsu insisted that the spent fuel storage period should be limited to 50 years. They would not accept the idea of an indefinite storage period.

T. TANAKA (OECD/NEA): We have published a report that may be interesting in that connection — Learning and Adapting to Societal Requirements for Radioactive Waste Management.

J. BOUCHARD (France — Chairperson): I should like to make a comment about what Mr. Forsström called the ‘wait and see’ option, which is not an ideal option for the management of the back end of the nuclear fuel cycle but is, unfortunately, the only option for most countries.

In your presentation, Mr. Tanaka, you mentioned several times that the ‘wait and see’ option is not acceptable from the Joint Convention perspective. But in that case what actions are being taken to try to clarify what will be the consequences of the ‘wait and see’ option? Is some action being taken by OECD/NEA in this respect?

T. TANAKA (OECD/NEA): The ‘wait and see’ policy is not a solution. Many countries have decided to implement geological disposal as the end point, and we should endorse their approach.

H.G. FORSSTRÖM (IAEA): ‘Waiting and seeing’ does not mean that one is not doing anything. It means that one is not yet deciding what the next step should be.

In my view, the important thing is that, whatever the next step, we all agree that there will be a need for geological repositories. It is important that geological repositories be developed and fortunately there are some countries engaged in geological repository development efforts.

Even in a 'wait and see' situation, you need to have your options open and you need to work on those options.

MULTINATIONAL APPROACHES RELEVANT TO SPENT FUEL MANAGEMENT

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Abstract

The storage of spent fuel is a suitable candidate for a multilateral approach, primarily at the regional level. Small countries with only a few nuclear power plants would benefit economically from large joint facilities. The storage of special nuclear materials in a few safe and secure facilities would also enhance safeguards and physical protection. However, the final disposal of spent fuel and high level radioactive waste is the best candidate for a multilateral approach. It would offer major economic benefits and substantial non-proliferation benefits in spite of the legal, political and public acceptance challenges to be expected in most countries. The transfer of nuclear waste from the exporting country to the host country of an interim storage facility or of a final repository would be done under bilateral or multilateral agreements at the commercial and governmental levels, in accordance with the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Bilateral or international oversight of joint facilities should be arranged, as needed, to achieve the confidence of the partners as to the safety and physical security of the proposed facility. Such monitoring should cover the adequacy of the technical design, its safety features, its environmental impact, the physical security of nuclear materials and possibly the financial management of the joint venture. After the initial choice of bilateral arrangements, some kind of international monitoring may become appropriate. Various organizations could fulfil such a function, in particular, the IAEA. Such monitoring would have nothing to do with nuclear safeguards; repository monitoring would be a parallel but independent activity of the IAEA.

1. INTRODUCTION

Over the past decades the countries with operating nuclear power plants have attempted to develop domestic solutions for the disposal of radioactive

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waste, be it the high level waste resulting from chemical reprocessing of spent fuel at home or abroad or the direct disposal of spent nuclear fuel. These efforts have met with mixed success. Large nuclear countries such as the USA, the Russian Federation, France, the UK, Japan and Germany have not yet succeeded in bringing into operation suitable disposal facilities. Small countries, in particular Finland and Sweden, have been more successful in overcoming the technical and political hurdles to achieving the objective of a truly closed nuclear fuel cycle — 'uranium ore out of geological formations, waste back into geological repositories'. In most countries, high level waste and spent fuel are simply stored temporarily in surface facilities awaiting solutions of a more permanent nature. Therefore, the interim storage of waste (whether separated high level waste or spent fuel) has become a necessary and crucial prerequisite to their final disposal.

The management of spent fuel falls under the primary responsibility of national nuclear operators and of their own governments. Nevertheless, multilateral and multinational approaches have received more and more attention in recent years ([1], para. 304); indeed they deserve serious consideration at all steps of spent fuel management, starting with interim storage facilities and ending with final geological disposal.

Interim storage facilities are in operation and are being built in many countries, at the reactor site or as a central national facility. There is no international market for services in this area, except for the service offered by the Russian Federation in receiving Russian origin spent fuel from Russian supplied power plants in Northern and Eastern Europe — with the possible offer to also do this for other spent nuclear fuel of non-Russian origin at some future date. The Russian approach leaves a lot of ambiguity as to the border line between the definitions of 'intermediate' storage and 'long term' disposal.

The storage of spent fuel could be a candidate for a multilateral approach, primarily at the regional level. Small countries with only a few nuclear power plants would benefit economically from larger joint facilities. The storage of special nuclear materials in a few safe and secure facilities would also enhance safeguards and physical protection.

(a) (b)

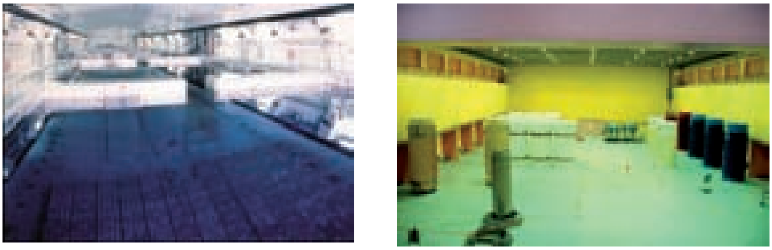

FIG. 1. Central wet storage in Sweden (a) and central dry storage in Switzerland (b).

The difficulties caused by national legislations in relation to the import of radioactive waste could, of course, discourage the establishment of joint multinational facilities in spite of the interim nature of such storage. The final disposal of spent fuel and high level radioactive waste is a better candidate for a multilateral approach. It offers major economic benefits and substantial non-proliferation benefits, in spite of the legal, political and public acceptance challenges to be expected in most countries if waste were to be imported ([1], (para. 301). From the technical and environmental standpoint, the international option would allow the choice of geological site characteristics across several countries.

2. WHICH ARE THE POTENTIAL HOST COUNTRIES FOR MULTINATIONAL GEOLOGICAL DISPOSAL?

The Russian Federation was the first country to express and formulate its willingness to receive foreign nuclear waste — in some detail but also with some ambiguity. Are there other countries? First on the list should be countries with favourable conditions, with very stable geologies and with vast areas. Australia is a prime example. However, a reluctance of national governments

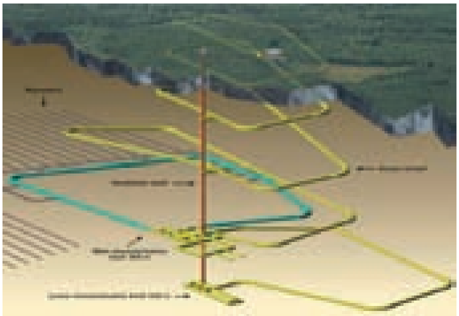

FIG. 2. Final disposal in Finland.

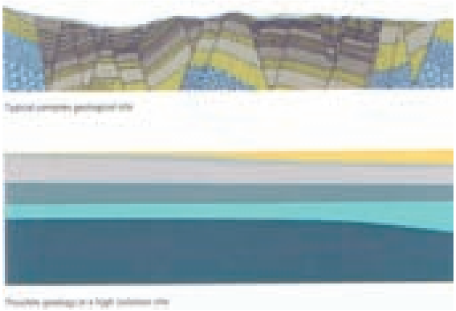

FIG. 3. Geology in Europe (above) and in Australia (below).

to import foreign radioactive waste precludes the selection of a site solely on the basis of technical and safety arguments. In the context of 'fresh fuel lease-spent fuel take back' arrangements promoted in particular by the USA, an engagement of other large nuclear countries in that discussion would be welcome in terms of economics for the customer countries and in terms of non-proliferation for the world community.

The Russian Federation has stated its interest in storing spent fuel on a long term temporary basis, a proposal that could possibly be extended to disposal later. The USA, already confronted with major public opposition to the repatriation of HEU fuel from the research reactors exported by its companies over the last decades, has expressed no interest in storing or disposing of foreign fuel. Yet in view of the somewhat exaggerated concerns expressed by its politicians and pundits over the risks of the back end of the nuclear fuel cycle, the USA should volunteer to give shelter to the spent fuel of a world which contains the nuclear technology and uranium of the USA. A few years ago, the large UK company BNFL tried to develop a solution which would have allowed it to complete its nuclear services (from cradle to grave) over the whole fuel cycle through a partnership with Australia, the country with the best geological sites in the world for ultimate disposal.

3. WHO ARE THE POTENTIAL CUSTOMERS?

In this context, 'customer' is understood to mean countries that would ship nuclear waste beyond their own borders for storage or disposal. It can be assumed as a premise that there would be no export from large countries with large domestic nuclear power programmes (nor from nuclear weapons states). The potential customers would first of all be small countries with a few nuclear power plants, with or without a suitable geology but seeking economical solutions by grouping resources. Countries with sizable nuclear power programmes but with few suitable geological sites, e.g. Japan, would also be possible customers.

4. NO 'SHIP AND FORGET'

The transfers of nuclear waste from an exporting country to the host country of an interim storage facility or of a final repository would be done under various bilateral or multilateral agreements at the commercial and governmental levels. All participating countries would presumably be signatories of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (the Joint Convention), the major legal instrument existing in the field [2].

The Joint Convention applies to spent fuel and radioactive waste resulting from civilian nuclear reactors and applications and to spent fuel and radioactive waste from military or defence programmes, if and when such materials are transferred permanently to and managed within exclusively civilian programmes, or when declared as spent fuel or radioactive waste for the purpose of the Convention by the Contracting Party. The Convention also applies to planned and controlled releases into the environment of liquid or gaseous radioactive materials from regulated nuclear facilities.

The obligations of the Contracting Parties with respect to the safety of spent fuel and radioactive waste management are based to a large extent on the principles contained in the IAEA's Safety Fundamentals publication on The Principles of Radioactive Waste Management [3], published in 1995. They include, in particular, the obligation to establish and maintain a legislative and regulatory framework to govern the safety of spent fuel and radioactive waste management, and the obligation to ensure that individuals, society at large and the environment are adequately protected against radiological and other hazards, inter alia, by appropriate siting, design and construction of facilities and by making provisions for ensuring the safety of facilities both during their operation and after their closure. The Joint Convention imposes obligations on Contracting Parties in relation to the transboundary movement of spent fuel and radioactive waste, based on the concepts contained in the IAEA Code of Practice on the International Transboundary Movement of Radioactive Waste [4].

The Joint Convention addresses the international aspects of waste management, e.g.:

Preamble (xi)
"Convinced that radioactive waste should, as far as is compatible with the safety of the management of such material, be disposed of in the State in which it was generated, whilst recognising that, in certain circumstances, safe and efficient management of spent fuel and radioactive waste might be fostered through agreements among Contracting Parties to use facilities in one of them for the benefit of the other Parties, particularly where waste originates from joint projects"

Article 1 — Objectives
"The objectives of this Convention are:
(i) to achieve and maintain a high level of safety worldwide in spent fuel and radioactive waste management, through the enhancement of national

measures and international co-operation, including where appropriate, safety related technical co-operation;"

The Joint Convention does not envisage an international verification system to ensure that national waste facilities respect the safety requirements spelled out in the Convention, whether or not a national facility contains foreign waste. However, it may be predicted that facilities containing foreign waste would be verified to some degree by the exporting countries. For domestic and international political reasons there would be a need for some kind of monitoring of spent fuel storage and disposal (even waste) after shipment, to provide political protection of the exporting country from accusations such as 'irresponsible dumping'. It is clear that an international waste management solution would not be of a simple commercial nature along the lines of 'ship and forget'.

The IAEA Expert Group on Multilateral Approaches, brought together by the IAEA's Director General in 2004–2005, took a serious look at this matter and it supported the principle of multilateral storage and disposal arrangements:

"The IAEA could facilitate this arrangement by acting as a 'technical inspection agency' assuring the suitability of the facility and applying state-of-the-art safeguards control and inspections". ([1], para. 304)

"It is also important that international oversight of an MNA be arranged, as needed, to achieve confidence of partners on adequate safety and physical security of the proposed facility". ([1], para. 339)

In several cases, domestic policy in the customer's State would require explicit assurances that the transferred waste will be properly managed and not simply dumped at some distant site. This would, in particular, be the case for Switzerland. The new Swiss Nuclear Law, which came into force in February 2005, addresses the issue in its Article 33:

"A permit will be granted for the export of nuclear waste…, when the following conditions are fulfilled…:

The recipient State has approved the import of nuclear waste under a government-to-government agreement;

A suitable nuclear installation is available in the recipient State satisfying up-to-date scientific and technical standards;

Transit States have approved such transports;

The sender has firmly agreed with the recipient of nuclear waste — with the endorsement of the authority designated by the Swiss Government — that such waste may be returned to the sender in case of necessity."

Should Switzerland choose to export waste, the second condition above would clearly oblige the Federal Government to ascertain, in one way or the other, that the installation is and remains 'suitable' and that it will satisfy state of the art technical requirements and standards.

What is remarkable in this law is the mirror clause applying to the import of nuclear waste. Import will indeed be possible, under the same conditions as spelled out in Article 33. This is not too surprising. With a strong chemical industry, Switzerland has long experience in bi-directional international transfers of toxic waste, with the routine import and export of various kinds of waste, and in ensuring the optimization and specialization of disposal facilities. All such transfers occur under the stringent regulations of the international Basel Convention, with special rules applying to transfers within the OECD/NEA [5].

5. FROM BILATERAL TO INTERNATIONAL MONITORING THROUGH THE IAEA

At first, the parties — sender and recipient — would agree on some kind of bilateral monitoring by dedicated bilateral teams or through international commercial companies that would provide technical services focusing on quality, environmental, health, safety, social accountability and information management issues, such as 'Bureau Veritas' or 'Société Générale de Surveillance'. A possible minor initial role might also be given to the IAEA in such schemes to add an international confidence level.

At a later stage, after the establishment of many bilateral arrangements, some kind of international monitoring may become appropriate. Various organizations could fulfil such a function, in particular the IAEA.

What kind of monitoring? Or, rather "What kind of assurances are to be provided?"

First of all, it should be clearly understood that such monitoring would have nothing to do with nuclear safeguards, that is, with the non-proliferation mandate of the IAEA. Nuclear safeguards would be a parallel and independent activity of the IAEA. In any case, the providing State would be of no proliferation concern in the cases of the Russian Federation and the USA, as declared nuclear weapons States. In non-nuclear weapons states, e.g. Australia, normal safeguards would apply automatically and would provide satisfactory non-proliferation assurances. Nonetheless, in the case of the nuclear weapons States the supplying countries would certainly want to ensure that nuclear waste transferred under a storage/disposal agreement would not be diverted to the host country's weapons programme.

The monitoring of the IAEA could deal with the following areas, with a scope depending on the bilateral transfer agreement concluded between the parties:

Technical design — proper international design standards. In line with the Joint Convention, the facility would have to satisfy international state of the art design norms, as well as technical and quality assurance standards. Customer countries may wish to give the IAEA more or less authority in this context.
Safety — design and operation to exclude accidents. The design, maintenance and operating procedures of the facility should exclude the possibility of nuclear accidents. In this context there would be review by the IAEA of national enforcement mechanisms.
Environmental — design and operation to exclude environmental damage. The design, maintenance and operating procedures of the facility should exclude the possibility of radioactive contamination of the environment above a certain limit for the whole operational life of the facility, in accordance with domestic and international norms.
Security — design and operation to exclude misuse and thefts of nuclear materials. The formulation of the Euratom Treaty, Article 77, is relevant for this task:

> "In accordance with the provisions of this Chapter, the Commission shall satisfy itself that, in territories of Member States, a) ores, source materials and special fissile materials are not diverted from their intended uses as declared by the users." When applied to a joint international facility, this would read: "The IAEA shall satisfy itself that, in the storage or disposal facility, spent fuel and other materials are not diverted from their intended uses as declared by the users".

Under this heading, the physical protection of the nuclear materials should be fully implemented in line with the IAEA defined guidelines (INFCIRC-225) [6].

Financial management — sound use of invested resources, especially in the case of joint financing of facilities. Different models are possible, e.g. financial matters only in the hands of the recipient country (with no monitoring), or a joint trusteeship (in which case the monitoring could be bilateral, or even with a third party at the table, e.g. the IAEA).

6. INSTITUTIONAL INTERGOVERNMENTAL ARRANGEMENTS

There would be a need for a solid nuclear legal and regulatory basis in the recipient country. IAEA publications include many details about the requirements for setting up a firm legal basis that would create the necessary trust of the international partners [7]. As already noted, the intergovernmental arrangements should refer specifically to international legal instruments in order to help create a smooth legal overlap between giver and recipient countries, e.g. the Joint Convention referred to above.

A number of detailed legal questions would need to be settled between the partners, such as the long term liability (host country or shared?) and the ultimate ownership of the nuclear waste (host country or providers?), especially in relation to the retrievability of buried nuclear waste, a decisive factor for some countries as exemplified above by the last paragraph of the Swiss Nuclear Law. Retrievability may be required for political, safety or environmental reasons, but also for the possible recovery of plutonium fuel in the event of future uranium shortages.

7. BUSINESS AND COMMERCIAL MODEL

There is also a need for a solid business and commercial basis between the partners, to make it clear who is responsible for providing the services and financial contributions necessary to ensure the smooth operation of the partnership, e.g.:

(a) A trusteeship of recipient countries and providers;
(b) Joint definition of the kind of services expected from the IAEA;
(c) Monitoring services to be paid to the IAEA on a time and expenses basis.

The IAEA would have to establish a ‘monitoring model’ (the IAEA could refuse to engage if the scope were incomplete, since the IAEA’s reputation would be at stake). The IAEA would report comprehensively to the trusteeship on a yearly basis, and succinctly in its own Annual Report.

8. IAEA ADMINISTRATIVE MODEL

How would the IAEA organize such monitoring work internally? Once again, it should be emphasized that such an activity has no relationship to nuclear safeguards; these would not be safeguards inspections. Therefore, the work would not be entrusted to the Department of Safeguards of the IAEA, but rather jointly to the Department of Nuclear Energy and the Department of Nuclear Safety and Security. Here the IAEA would be performing a service to its Member States.

In practical terms the IAEA would set up ad-hoc internal teams, with personnel drawn from these two Departments, with the occasional involvement on a personal basis of some safeguards inspectors and with safeguards technical support. Because of the required confidentiality, this would not be a team of external experts, as used by the IAEA for peer review missions in the safety and nuclear licensing fields.

As far as practical verification arrangements are concerned, the IAEA would make use of human and technical resources to carry out its monitoring functions. There would be human inspections with the physical and visual review of facility features, the taking of environmental samples to assess possible leaks and spills, etc. The technical equipment in support of inspections would include radiation detection equipment, seals and sampling equipment. In special situations, the IAEA could also call on remote monitoring, that is, with tamperproof digital cameras transmitting pictures back to IAEA headquarters on a regular basis or upon image changes.

The IAEA would have to report in an appropriate fashion on the findings of its verification activities. Upon detection of irregularities this would be done as soon as possible to the partners, on a confidential basis. At yearly intervals the IAEA would submit to the partners a confidential Annual Report. However, by its very status as an independent international organization, the IAEA would need to report briefly at yearly intervals to its own constituency, the Board of Governors, on the general scope of the controls performed.

9. CONCLUSIONS

The final disposal of spent fuel (and radioactive waste) in shared repositories must be looked upon as only one element of a broader strategy of parallel options. National solutions should remain a first priority. This is the only approach for States with many nuclear power plants in operation or in past operation. For other States with smaller civilian nuclear programmes, a dual track approach may be more appropriate in which both national and international solutions are pursued. Small countries should keep options open (national, regional or international), if only to maintain the minimum national technical competence necessary to act in an international context ([1], para. 302).

On 1 June 2006, the international Weapons of Mass Destruction Commission chaired by former IAEA Director General Hans Blix handed over its report to the Secretary General of the United Nations, Kofi Annan [8]. The report notes:

> "The IAEA has long served as a forum for considering proposals relating to the fuel cycle and for new types of nuclear power reactors. It is desirable that States continue to use the IAEA for these purposes, e.g. to discuss the ideas of fuel banks, regional arrangements for the production of fuel, and the management and disposal of spent fuel, as well as the possibility of proliferation-resistant fuel cycles."

As far as the management and disposal of spent fuel is concerned this is, first of all, a plea for multilateral and international approaches formulated from a non-proliferation perspective. Yet, non-proliferation will hardly be the driving force since international safeguards already exist for that purpose and since the risk associated with spent fuel is quite limited, in particular for recycled spent fuel.

The driving forces towards multilateral facilities for the back end of the fuel cycle will be economics and practical considerations. Once the political burden of public acceptance has been overcome, the benefits of combining facilities will be overwhelming. While still not politically acceptable, one can argue with a fair degree of confidence that the 27 country European Union will not build 16 deep repositories in each of the countries operating a nuclear power plant. Logic and common sense will ultimately prevail in Europe and elsewhere in the world.

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DISCUSSION

A. GONZÁLEZ (Argentina): Perhaps the most important problems connected with the multinational approaches relevant to spent fuel management are legal problems.

For example, studies have shown that parts of Patagonia are eminently suitable for the storage of radioactive waste, but the import of radioactive waste into Argentina is prohibited by law. The same applies to the Russian Federation, for example where legal problems have arisen even in connection with the import of small spent sources.

Maybe the IAEA could establish a forum for the consideration of such legal problems with a view to solving them, but I am not very optimistic about the prospects for success.

B. PELLAUD (Switzerland): There are national laws — even local ones — prohibiting the importation of radioactive waste, but one can solve the legal problems by changing the laws. Of course, that requires political decisions.

In the light of the adoption, in 1989, of the Basel Convention on the Control of Transboundary Movements of Hazardous Wastes and their Disposal, which arose out of scandals about the dumping of toxic waste in African countries, the OECD/NEA member countries committed themselves to transferring toxic waste only among themselves. Perhaps arrangements like those arrived at by various OECD/NEA member countries in respect of toxic waste could be arrived at for spent fuel, although it might take as long as 50 years. Much depends on whether or not there will be an increased acceptance of nuclear power generation.

H.G. FORSSTRÖM (IAEA): Laws are a reflection of the political situation, and the political situation is a reflection of public opinion. What developments could, in Mr. Pellaud's view, change the political situation in favour of the establishment of multinational facilities for the management of spent fuel?

B. PELLAUD (Switzerland): In my view, if there is a major expansion of nuclear power (which, in most of the countries in question, would happen with the backing of the public), there will occur a kind of 'crystallization' among politicians, who will start asking, "Couldn't we do things better by not having about 50 enrichment plants, 10–20 reprocessing facilities and 30–50 geological repositories?"

As regards developments at the grass roots level, I would mention that in my country the socialists are in favour of Switzerland's joining the European Union but not of the establishment of multinational facilities for radioactive waste management; they believe that every country should manage its own radioactive waste. However, the younger generation is concerned less about radioactive waste than about global warming, so things may change — and I think that applies to other countries as well.

T. TANIGUCHI (IAEA): Mr. Pellaud mentioned the monitoring of good management, of the absence of dangers to humans and the environment, and of good physical protection and material security. In my view, the absence of dangers to humans and the environment is exactly what safety is. Also, good management is, together with sufficient financial resources, a precondition for safety. In that sense such monitoring would basically cover safety and security. I wonder if there is anything beyond safety and security, because how to monitor safety and security is a challenging issue when it comes to multinational facilities.

Safety is covered by the Joint Convention and the Safety Standards, and physical protection and security is covered by the amended Convention on the Physical Protection of Nuclear Material. We are now developing basic guidance reports for security. In that sense, the international framework in this area is evolving, but I feel that to go beyond peer reviews to monitoring is a sensitive issue for the IAEA.

B. PELLAUD (Switzerland): I see no difference here between a multinational facility and a national facility. In neither case will the technical design be subjected to international scrutiny, because the national safety authority will decide what is appropriate.

Safety would be a national responsibility, with the IAEA carrying out peer reviews of the kind it is already carrying out at national facilities in order to assess if the safety is being properly handled.

One could imagine a degree of environmental monitoring where releases of water or gas are collected simply to make sure that there is no excess contamination. That also has been done in one form or another by the IAEA.

As regards security, rules exist. There are standards, for example INFCIRC/225/Rev. 4 (Corrected), covering the storage of nuclear material, including spent fuel.

It would involve several departments of the IAEA because it concerns waste on one side and safety on the other. But I think all the mechanisms are in place. They would simply need to be 'repackaged', so to speak, re-described in such a way that they could immediately be recognized as applicable to a national and a multinational facility.

A. GONZÁLEZ (Argentina): I agree with Mr. Taniguchi, and I am not convinced by what Mr. Pellaud has just said.

The problem is that, if international monitoring occurred, it could occur only in respect of issues where countries have entered into legally binding commitments, and this has happened only in the safety area. There are no legal undertakings in the area of financial management (countries can do whatever they want with their financial resources), nor in the area of technical design (countries can design facilities in any way they like). The only legally binding undertaking of countries in the spent fuel safety area is the Joint Convention. That can be monitored, but as to the rest I think it is very unlikely that countries will agree to be monitored.

B. PELLAUD (Switzerland): There is a misunderstanding here which will be visible in the written paper. When I talked about international monitoring, I was not talking about the international community as a whole passing judgement on the way in which a multilateral facility is operated. There would be a multilateral agreement among the governments involved, which would decide at some point in time that they will entrust the monitoring to a particular body — which could be a private company like Bureau Veritas or the IAEA. The IAEA would not carry out monitoring pursuant to a convention, but as a service.

The real question is "Can the IAEA perform such services? Would it be appropriate?"

T. TANIGUCHI (IAEA): On this important issue it is very clear that, when a multinational facility is established, it should be under the responsibility of the regulatory authority of the country where it is located.

GLOBAL NUCLEAR ENERGY PARTNERSHIP

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Abstract

The paper describes the USA's initiative called the Global Nuclear Energy Partnership which, amongst other things, seeks to reduce the generation of radioactive waste from the nuclear fuel cycle and to minimize the associated risks of nuclear proliferation. The paper outlines the reasons for the initiative and its essential technical aims.

Last year, US Secretary of Energy Sam Bodman called senior staff together and told us that President Bush had called for a significant expansion of nuclear power. The Secretary formed a task force under Deputy Secretary Sell, and challenged us to do three things:

(1) Review and make recommendations on additional policy options to support the Department's spent fuel management obligations;
(2) Review and evaluate all aspects of our domestic and international policies related to the civilian use of nuclear power;
(3) Develop and recommend policies that would promote the use of nuclear power in the USA and internationally as a means of safe, pollution free, environmentally sound and plentiful energy.

The result is the initiative that was named the Global Nuclear Energy Partnership, which we also call GNEP, and which President Bush announced last February.[1]

I would like to tell you today why the USA is proposing GNEP and to elaborate on exactly what it is and how we propose to get started. The President set a policy goal of promoting a great expansion of nuclear power both in the USA and around the world. There are many important reasons for this. I will focus my remarks on the most important.

The Department of Energy projects that total world energy demand will double by 2050. Looking only at electricity, the projections indicate that there will be a 75% increase in electricity demand over the next 20 years. Nuclear power is the only mature technology capable of providing large amounts of completely emission-free base load power to meet this need. Expanding its use would result in significant benefits around the globe: reduced world greenhouse gas intensities, pollution abatement, and the security that comes from greater energy diversity.

But nuclear power, with all its potential for benefiting humanity, carries with it two significant challenges. First, what should be done with the nuclear waste? Second, how can the proliferation of fuel cycle technologies that can lead to the development of nuclear weapons be avoided?

GNEP seeks to address and minimize these two challenges by developing technologies to recycle the spent fuel in a more proliferation resistant manner, and by supporting a re-ordering of the global nuclear enterprise to encourage new fuel cycle technologies, credible fuel supply assurances and strong commercial incentives for leasing nuclear fuel, instead of building indigenous enrichment and reprocessing capabilities.

Regarding our own policy on spent nuclear fuel, the USA stopped the old form of reprocessing in the 1970s, principally because of heightened domestic concerns about separated plutonium production. But in other major nuclear economies, for example, France, Japan, the United Kingdom, the Russian Federation, reprocessing continued and expanded.

The world today has a total of nearly 250 t of separated civilian plutonium. That is 250 000 kg. The IAEA says that a nuclear weapon can be made with 8 kg of separated plutonium. That should concern us all.

Each year, the world's nuclear power plants produce roughly 10 000 t of spent fuel. About 1% of this spent fuel is a mixture of plutonium and other fissionable isotopes. In the USA, about 2300 t of spent fuel are produced each year. Applying the rough 'rule of thumb' that each ton of fissionable material can produce 1 GW of electricity, 23 GW of electricity could be produced from the fissionable material in US spent nuclear fuel each year.

Although the USA foreswore reprocessing, the world's reprocessing capacity has continued to expand. There is now almost enough capacity to reprocess almost half the world's output of spent fuel annually.

If we consider only the USA we are on the verge of a nuclear renaissance. In many respects this is due to the provisions enacted in the Energy Policy Act of 2005. New plants will be built, but if many more are to be built the USA must rethink the wisdom of its once through spent fuel policy. With nearly one fourth of the world's nuclear capacity, the USA must move to a waste management system that recycles spent fuel.

Unless a technological or policy alternative is provided, as the use of nuclear power expands around the globe other nations will find it commercially attractive to address the buildup of spent fuel with reprocessing. In doing this they will inherit the burden of safeguarding and protecting the technologies and nuclear materials that reprocessing produces. It also means that there will be multiple waste repositories around the globe. A fully functioning repository is required whether or not fuel is reprocessed. But if the nations of the world work together, using the economies of scale, the growth of the number of enrichment, reprocessing and repository facilities can be restrained — to everyone's benefit.

The USA remains confident that Yucca Mountain is the best location for a permanent geological repository in the USA. Licensing and opening the Yucca Mountain facility remains a top priority. Whether or not the USA recycles, the Yucca Mountain repository is needed. However, the statutory capacity limit of the Yucca Mountain repository, as currently configured, will be exceeded by 2010. If nuclear power's contribution to US electricity demand remains only at 20% for the rest of the century, it will be necessary to build the equivalent of nine Yucca Mountain repositories to contain 'once through' spent fuel.

The current US Administration believes that the wiser course is to recycle the used fuel coming out of the reactors, thereby reducing its amount and its radiotoxicity, so that only one Yucca Mountain equivalent would be required for the rest of this century. It believes that a system must be built that reaches equilibrium with the production of spent fuel — so that inventories of plutonium are consumed to produce power — and destroyed so they cannot fall into the wrong hands. If this technology and policy solution is right for the USA, then it should also be made available to other States through mutually acceptable, commercially attractive arrangements.

So what exactly is GNEP? GNEP is really about identifying the policies, developing the technologies and building the international regimes that would manage and promote a growth in nuclear electricity generation that enhances common waste management and non-proliferation objectives. The initial programme would focus on a few of the important engineering and development efforts that are key to the success of GNEP.

First, the Department of Energy seeks to greatly accelerate its work on the demonstration of advanced recycling. This effort would build on the advanced fuel cycle initiative started by the US Congress several years ago.

In the laboratory, recycling technology has been developed that does not separate plutonium, unlike the current reprocessing technologies used around the globe. Rather, this process keeps the actinides together, including plutonium, so that they can be made into fuel to be consumed in fast reactors that will also produce electricity. By not separating plutonium and building the most advanced safeguards technologies into the system, recycling can be performed in a way that greatly reduces proliferation concerns.

Another key objective of GNEP is to demonstrate, on an engineering scale, an advanced burner reactor that will consume plutonium and other actinides, extracting energy potential from recycled fuel, reducing the radiotoxicity of the waste in repeated cycles so that the resultant waste requires dramatically less geological repository space and is in an environmentally robust form.

These technologies would be bound together in a ‘reliable fuel services’ framework. GNEP would build and strengthen a reliable international fuel service consortium under which fuel suppliers who chose to operate both nuclear power plants and fuel production and handling facilities would provide reliable fuel services to States that choose to only operate nuclear power plants. This international consortium is a critical component of the GNEP initiative because it would provide a strong, commercially attractive alternative to the spread of enrichment and reprocessing technologies.

In exchange for an assured fuel supply on attractive commercial terms, States that are interested in bringing the benefits of nuclear power to their economies would have a strong commercial and policy incentive not to invest in enrichment and recycling. The Non-Proliferation Treaty does not restrict the right of States to build and operate technologies that have the potential capability of producing materials usable for creating weapons. States have a sovereign right to build and deploy enrichment and recycling facilities. However, a compelling commercial alternative could discourage them from acting on those rights. Participating in a mutually advantageous global nuclear enterprise would enhance energy and physical security.

There are two other key elements of GNEP requiring further technology development. The USA would hope to work in partnership with other States to develop exportable, proliferation resistant, perhaps modular or factory built, reactors that are appropriate to the needs of the developing world. In all cases we would work to develop and incorporate the most advanced safeguards technologies, and ensure and emphasize best practices for the handling of nuclear materials worldwide.

So, what is the status of the initiative? In the fiscal years 2006 and 2007 the Department of Energy proposed concentrating its efforts on technology development to support a 2008 decision on whether to proceed with these demonstrations. In general terms, the Department’s US $250 million request for funding in 2007 is to initiate work on separation and advanced fuels technology development, transmutation engineering, systems analyses and planning functions to support the development of an advanced spent fuel recycling plant, and to support the demonstration of a commercially deployable reactor that consumes actinide based fuels.

In conclusion, all available technologies should be pursued to address the anticipated growth in the demand for energy. But the growth of nuclear energy is vitally important for the USA and for the world. Our country has chosen to change its current path and to begin the transformation to a new, safer and more secure approach to nuclear energy — an approach that brings the benefits of nuclear energy to the world while reducing vulnerabilities from proliferation and from nuclear waste.

Challenges remain in demonstrating the GNEP technologies. But without GNEP the inventories of plutonium throughout the world will continue to increase for generations to come. There will be a greater volume of high level radioactive waste, and its radiotoxicity will not be reduced. There will be a greater proliferation risk. There will be greater amounts of greenhouse gases emitted into the environment and less energy both in the USA and abroad.

The Global Nuclear Energy Partnership is not a 'silver bullet', but it is part of a broad strategy that, when combined with advances in renewable energy sources, clean coal and other technology developments, can and will make a difference to the security, environmental and energy challenges that we face. More information can be found at the US Department of Energy's web site: www.GNEP.gov.

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[1] **President Bush's Radio Address Focuses on Energy Issues: WASHINGTON, DC** — This morning President Bush discussed the Global Nuclear Energy Partnership (GNEP) during his weekly radio address. The transcript is below.

THE WHITE HOUSE
Office of the Press Secretary
(Lake Buena Vista, Florida)

At 10:06 A.M. EST
Saturday, February 18, 2006
RADIO ADDRESS BY THE PRESIDENT
TO THE NATION

THE PRESIDENT: Good morning. This coming week, I will visit Wisconsin, Michigan, and Colorado, to discuss our strategy to ensure that America has affordable, reliable, and secure sources of energy. The best way to meet our growing energy needs is through advances in technology. So in my State of the Union Address, I announced the Advanced Energy Initiative. We will pursue promising technologies that will transform how we power our vehicles, businesses, and homes — so we can reduce our Nation's dependence on foreign sources of energy.

This morning, I want to speak to you about one part of this initiative: our plans to expand the use of safe and clean nuclear power. Nuclear power generates large amounts of low-cost electricity without emitting air pollution or greenhouse gases. Yet nuclear power now produces only about 20 per cent of America's electricity. It has the potential to play an even greater role. For example, over the past three decades, France has built

58 nuclear power plants and now gets more than 78 per cent of its electricity from nuclear power. Yet here in America, we have not ordered a new nuclear power plant since the 1970s. So last summer I signed energy legislation that offered incentives to encourage the building of new nuclear plants in America. Our goal is to start the construction of new nuclear power plants by the end of this decade.

As America and other nations build more nuclear power plants, we must work together to address 2 challenges: We must dispose of nuclear waste safely, and we must keep nuclear technology and material out of the hands of terrorist networks and terrorist states.

To meet these challenges, my Administration has announced a bold new proposal called the Global Nuclear Energy Partnership. Under this partnership, America will work with nations that have advanced civilian nuclear energy programs, such as France, Japan, and the Russian Federation. Together, we will develop and deploy innovative, advanced reactors and new methods to recycle spent nuclear fuel. This will allow us to produce more energy, while dramatically reducing the amount of nuclear waste and eliminating the nuclear byproducts that unstable regimes or terrorists could use to make weapons.

As these technologies are developed, we will work with our partners to help developing countries meet their growing energy needs by providing them with small scale reactors that will be secure and cost effective. We will also ensure that these developing nations have a reliable nuclear fuel supply. In exchange, these countries would agree to use nuclear power only for civilian purposes and forego uranium enrichment and reprocessing activities that can be used to develop nuclear weapons. My new budget includes $250 million to launch this initiative. By working with other nations under the Global Nuclear Energy Partnership, we can provide the cheap, safe, and clean energy that growing economies need, while reducing the risk of nuclear proliferation.

As we expand our use of nuclear power, we're also pursuing a broader strategy to meet our energy needs. We're investing in technologies like solar and wind power and clean coal to power our homes and businesses. We're also investing in new car technologies like plug-in hybrid cars and in alternative fuels for automobiles like ethanol and biodiesel.

Transforming our energy supply will demand creativity and determination, and America has these qualities in abundance. Our Nation will continue to lead the world in innovation and technology. And by building a global partnership to spread the benefits of nuclear power, we'll create a safer, cleaner, and more prosperous world for future generations.

Thank you for listening. END

DISCUSSION

Y.A. SOKOLOV (IAEA): I see some controversy here, because it looks like you are inviting for partnership only countries with developed technology. There are no points at which the concerns or interests of developing countries can be addressed. If you are talking about ‘partnership’, both sides have to be included.

L. BROWN (United States of America): That is an important point. When you sit down with potential partners everybody brings capabilities to the table. To use an analogy, you may change tyres while the other party fixes radios, so the two of you come together in order to set up an automobile repair company. There are about 40 countries in the Nuclear Suppliers Group with different capabilities. Some build reactor vessels (which we do not do in the USA any more — we import them), some make nuclear instrumentation, and so on.

If there really is a renaissance of nuclear power, it will benefit everybody who is involved in some aspect of the nuclear business — whether it is uranium mining in Australia or reactor vessel manufacturing in the Republic of Korea. Where a country simply wants the nuclear energy in order, say, to reduce its dependence on oil or coal, we think it would welcome a multinational arrangement for the leasing of the nuclear fuel, which would be returned after use to the supplier — an arrangement rather like the one between the Soviet Union and some other countries of eastern Europe in the 1970s and 1980s.

A. GONZÁLEZ (Argentina): Mr. Brown’s point about restricting the spread of enrichment and reprocessing could give rise to controversy if a country feels that it is being restricted. If a country feels that it belongs to ‘the club’ and is therefore not being restricted, there will be no controversy.

Perhaps one could formulate the second point in a positive manner by talking not about ‘restricting’ something but about ‘ensuring the availability of’ something. That is perhaps the only area where a consensus would be necessary.

H.G. FORSSTRÖM (IAEA): I am very positive towards what you are saying, but I am worried sometimes that you are over-selling. Clearly you are trying to solve a radioactive waste problem for the USA. However, what you are not mentioning — and I think you must be honest and mention it — is that whatever you do will require a lot of recycling, and recycling means waste as there are always process losses. I think that for the sake of honesty it is important not to over-sell the long term waste aspect, because you will still have long term waste — perhaps at a level that is 5% instead of 100%, but you still have it there.

L. BROWN (United States of America): That is absolutely right; there are a lot of complexities to this idea.

We wanted to come up with something that we could show would actually work now. So we tried to take as much as possible ‘off the shelf’ and then focus on what the technology challenges would be. There may be some complexities, as you pointed out, that we have missed. However, one thing we said at the beginning was that, whatever we come up with, it will not be what we end up with. A lot of work has to be done, and it has to be done with a lot of detailed research, and it is going to take time.

J. BOUCHARD (France — Chairperson): We are very happy to see the USA rejoining the ‘closed fuel cycle club’. France is very interested in the proposals that have been made.

A. GONZÁLEZ (Argentina): After what happened about 20 years ago, what Mr. Brown has been saying is music to my ears. However, I would like to make some points.

One relates to — credibility. About 20 years ago the USA told us exactly the opposite of what it is telling us now, and I think it is necessary to be sure that what we are hearing now is national policy and that tomorrow we are not going to see an Administration making the pendulum swing to the other side again.

Also, I believe that for a partnership like the one envisaged by the USA to succeed there must be a clear commitment to an international nuclear safety regime. This is very important for the USA because, with the growing harmonization of safety standards, we perceive that the USA sometimes goes in a different direction. Even the units used for radiation safety purposes in the USA are different from those used in other parts of the world. So I believe that partnership needs an international nuclear safety regime, whatever form that regime takes, with harmonization as a precondition.

L. BROWN (United States of America): Regarding the first point, we recognize that a programme that may take 20 years — that is 10 Congresses, potentially five and at least three different Presidents, must succeed or fail on its own merits. Also, we think that the international community can play a role. If a strong partnership is developed in the remaining period of the current Administration and if a strong consensus is developed on what works globally, it will feed itself and survive future Administrations. If it becomes the norm rather than just a current policy, then we can make it survive.

J. BOUCHARD (France — Chairperson): I do not see any contradiction between the present position of the USA and the previous one. The objective remains the same.

INITIATIVE BY THE PRESIDENT OF THE RUSSIAN FEDERATION ON GLOBAL NUCLEAR INFRASTRUCTURE

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Abstract

At the beginning of 2006 the President of the Russian Federation promulgated an initiative inviting the international community to collaborate in the development of the Global Nuclear Infrastructure (GNI) in order to ensure equal access to nuclear energy by all interested countries while ensuring strict observance of non-proliferation obligations. Key elements of the GNI would be international centres that provided services related to the nuclear fuel cycle, including uranium enrichment and the management of spent nuclear fuel, as well as training and qualification centres for nuclear energy personnel from developing countries. The paper outlines the ideological basis of the initiative, the approach to the formation of a new technological platform for nuclear power in the Russian Federation, and the use of the GNI to provide an energy supply for globally sustainable development. The issues related to spent fuel management are highlighted.

1. PERSPECTIVES ON NUCLEAR POWER DEVELOPMENT IN THE RUSSIAN FEDERATION

Currently the Russian programme for the development of the energy sector is being revised with the objective of doubling annual electricity output by 2020.

In the Russian Federation, about 50% of the total electricity is generated by burning natural gas, about 18% by burning coal, 16% by nuclear power generation and 16% by hydroelectric power generation. The large share of gas in the total results in a decreasing reliability and stability of power supply for consumers. Further large scale use of gas in the power industry of the Russian Federation is complicated due to the increasing imbalance between the internal consumption and the export of gas. The Russian Federation may face a gas deficiency as early as the first quarter of this century. As a result, there is a need to have new nuclear power plants.

Currently, nuclear power in the Russian Federation is based on thermal reactors — WWER and RBMK types — which use uranium-235 as their main energy resource. The advantages of nuclear power as compared to fossil fuel are the following:

(a) The potential for export of commercial nuclear technologies, including WWER reactors and related technologies of the nuclear fuel cycle;
(b) A weak dependence on the fluctuations of natural uranium market prices;
(c) A negligible level of greenhouse and other hazardous gas releases;
(d) A low demand for transport infrastructure.

On the other hand, there are some disadvantages of nuclear power based on thermal reactors using the once-through fuel cycle:

(1) Incomplete utilization of the spent nuclear fuel management cycle;
(2) Unavoidable storage of continually increasing quantities of spent fuel (over 15 000 t of spent nuclear fuel have been accumulated to date, with a rate of accumulation of 800 t/a);
(3) Accumulation and storage of uranium enrichment tailings (the accumulation rate is about 4000 t/a);
(4) The need of a large capacity repository for radioactive waste;
(5) The limited resources of uranium-235 (see Fig. 1).

Uranium-235 resources in the Russian Federation are sufficient to achieve 25% of the total electricity generated by nuclear power plants by 2030, but this share would inevitably decrease substantially by the end of the century. At the same time, there are enormous strategic reserves of uranium-238, the energy potential of which cannot be used with current technology.

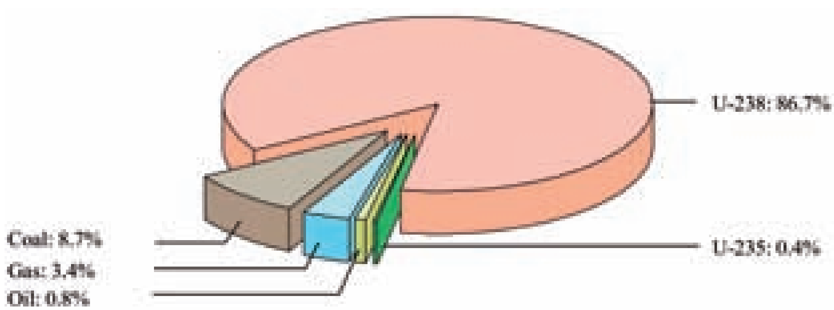

FIG. 1. Relative energy potential of natural resources of the Russian Federation.

As can be seen from Fig. 1, the energy potential of the uranium-238 resource is 10 times higher than that of coal and 25 times higher than that of natural gas.

With the objective of ensuring energy security for the Russian Federation, the nuclear power development programme envisions switching, by the middle of the 21st century, to a new technological platform based on the closure of the nuclear fuel cycle using fast reactors and, hence, the use of uranium-238.

The key technological components of innovative nuclear power have already been demonstrated at the commercial level in the Russian Federation:

(i) The BN-600 demonstration fast reactor with sodium coolant continues to operate successfully at the Beloyarsk nuclear power plant after 25 years;

(ii) The technology of mixed oxide fuel (MOX) fabrication for fast reactors has also been developed and demonstrated experimentally;

(iii) The technology for reprocessing of spent nuclear fuel from WWER-440 and BN-600 reactors is being successfully used at the RT-1 reprocessing plant, followed by the use of regenerated uranium in RBMK reactors and the storage of separated plutonium and vitrified radioactive waste.

It is expected that the transition from the present system to the innovative nuclear power system will be achieved in two phases: the preliminary phase (until 2030) and the final phase (2030–2050). During the preliminary phase the increase of nuclear capacity will be fulfilled primarily by the introduction of advanced WWER reactors. During the final phase, a further increase in nuclear capacity will be achieved primarily through the commissioning of commercial fast reactors.

The following steps have to be achieved during the transition period:

(a) Bringing the already demonstrated fast reactor and closed nuclear fuel cycle technologies to a commercial level, including:

— Commissioning the BN-800 fast reactor by 2012 as a pilot facility for utilizing MOX fuel and for testing technologies of the closed nuclear fuel cycle;
— Commissioning a commercial plant for radiochemical reprocessing of spent nuclear fuel from thermal reactors;
— Creating a commercial plant for the fabrication of MOX fuel for fast reactors using the plutonium extracted from reprocessed spent nuclear fuel from thermal reactors;
— Designing and constructing a small series (4–5 units) of first generation commercial fast reactors using MOX fuel;

(b) Research and development work to demonstrate an innovative reactor and associated fuel technologies, followed by their commercialization during the final phase of the changeover to innovative nuclear power.

In the future, the International Nuclear Centre will be established, providing services in the nuclear fuel cycle, including enrichment of uranium and management of spent fuel. By the end of the transition period, the technological infrastructure for innovative nuclear power will be created in the Russian Federation with the complete closure of the nuclear fuel cycle and the use of uranium-238 as the main raw material resource. In addition, the problems of the management of the spent nuclear fuel from thermal reactors will be resolved.

2. ORGANIZATIONAL AND TECHNOLOGICAL ASPECTS OF CREATING THE GLOBAL NUCLEAR INFRASTRUCTURE

In the nuclear States, considerable investments have been made and are being made in the development and commercialization of technologies of fuel enrichment, thermal reactors and spent nuclear fuel management. A significant number of the world's nuclear power plants are sited in the USA, France, the UK and the Russian Federation, and these countries are the main suppliers of nuclear power technologies and services. The large scale development of nuclear power is also planned in China and India in the next few decades.

As large scale nuclear energy is developed on a global scale there will be many economic, social, political, technological, environmental and non-proliferation challenges. It may be appropriate for the States possessing advanced technologies (firstly, the principal nuclear States) to take responsibility for finding ways to address these challenges.

It is a particular objective of the new initiative by the President of the Russian Federation to take this responsibility. The initiative arises from the established Russian national policy in this area. In particular, the Russian Federation provides a set of guaranteed commercial services in the area of nuclear power to all countries that strictly observe non-proliferation requirements. Russian assistance includes specialist training, sale and construction of nuclear power plants (the Russian WWERs), fresh fuel supply with the return of spent nuclear fuel to the Russian Federation, and the temporary storage of spent fuel from the foreign nuclear power plants in the Russian Federation. This creates opportunities for interested countries to have access to nuclear energy.

Additional opportunities were identified by the International Working Group on Multilateral Nuclear Approaches (MNA) established by M. ElBaradei, Director General of the IAEA. Such opportunities emerge as a result of the creation of international centres for providing services in the area of sensitive nuclear technologies, particularly uranium enrichment.

Because of the drawbacks of the nuclear power system based on the open fuel cycle and thermal reactors (low efficiency in the use of raw material resources and the accumulation of spent fuel), it is not capable of ensuring energy supplies for sustainable global development. This could, however, be achieved by a change to a new platform based on technologies of the closed fuel cycle using fast reactors. In this system, the recycling of plutonium together with uranium in fast reactors with a breeding ratio exceeding 1 makes it possible to fully utilize the energy potential of uranium-238, which is available in great amounts. In addition, this approach decreases the volumes of radioactive waste and facilitates its isolation.

The interest in fast reactors in the Russian Federation and in other countries that are considering the use of large scale nuclear power is based on the technological capabilities of reactors of this type to contribute significantly to creating a sustainable energy supply. The economic characteristics of nuclear power systems based on the closed fuel cycle and fast reactors have not yet been determined. On the other hand, it is obvious that the closure of the fuel cycle based on fast reactors within the framework of nuclear power at the national level would be uneconomical for most countries. The costs of development, demonstration and commercialization of fast reactors and related nuclear fuel cycle technologies are so immense that they can only be justified if they are used on a large scale. Estimates show that recycling technologies can be cost effective based on a rate of reprocessing of spent fuel from thermal reactors of at least 800 t/a. This is the spent fuel amount produced annually from systems based on WWER or PWR type reactors with a total installed power equal to 40 GW(e).

Under these conditions, many States in the world would be forced to continue to use thermal reactors and to delay decisions on spent fuel management. Ultimately, these countries would isolate spent fuel geologically, as this is the least complicated and least expensive option at the national level. However, this would be the least reasonable approach from the standpoint of supplying energy for global sustainable development. Instead of using plutonium accumulated in spent fuel to involve the vast resources of uranium-238 for energy production, the international community would have to organize perpetual control over numerous storage facilities and repositories of spent fuel containing many thousands of tonnes of plutonium.

Development of the Global Nuclear Infrastructure could be the alternative way to proceed. Within this alternative, an international fuel centre in the Russian Federation could provide services in the nuclear fuel cycle. From Global Nuclear Infrastructure perspectives, the most promising option would be fuel centres carrying out, under IAEA safeguards, the enrichment of uranium and the reprocessing of spent fuel from thermal reactors with further utilization of extracted uranium and plutonium in fast reactors.

3. EXAMPLES OF THE SERVICES OF AN INTERNATIONAL FUEL CENTRE (IN THE RUSSIAN FEDERATION) FOR URANIUM ENRICHMENT AND SPENT FUEL UTILIZATION FOR ENERGY PRODUCTION

In this context, the term 'international' refers to the centre functioning on the territory and under jurisdiction of the State supplying nuclear services and means, as follows:

(a) Providing services on a non-discriminatory basis of uranium enrichment and spent fuel management to all countries that strictly follow non-proliferation requirements;
(b) Controlling the activities of the centre by IAEA safeguards;
(c) Providing options for diverse cooperation within the framework of the international fuel centre for countries developing nuclear power on a large scale with fast reactors and closure of the nuclear fuel cycle;
(d) Providing the possibility for other countries to take part in the international fuel centre without access to sensitive technologies involving closed fuel cycles and fast reactors.

The international fuel centre covered by IAEA safeguards might provide the following technological services:

(1) Uranium enrichment, fabrication and shipment of fresh fuel for thermal reactors;
(2) Delivery and temporary storage of spent fuel from foreign thermal reactors;
(3) Reprocessing of spent fuel in a specially dedicated line at a reprocessing facility;
(4) Fabrication of fuel for fast reactors in a specially dedicated MOX fuel production line;

(5) Irradiation of MOX fuel for electricity generation in a fast reactor constructed specifically for this purpose;
(6) Interim storage of spent fuel from the fast reactor followed by its recycling in commercial fast reactors.

The establishment and operation of the first international fuel centre of this kind could enable the practical development of the principal technological and institutional solutions for a global nuclear infrastructure for the sustainable development of civilization. In particular, the work of this centre could demonstrate to the world community the possible ways to resolve the problem of accumulated spent fuel by the first quarter of this century, while preserving the non-proliferation regime.

In terms of non-proliferation, the centre's effectiveness would be ensured by the fact that a considerable fraction of civil plutonium accumulated in the world would be consolidated and utilized at the international fuel centre under IAEA supervision. Currently, this plutonium is stored as spent fuel in numerous storage facilities in many countries. The availability of centres providing services for spent fuel management on a non-discriminatory basis could eliminate the need for national programmes for the development of sensitive technologies.

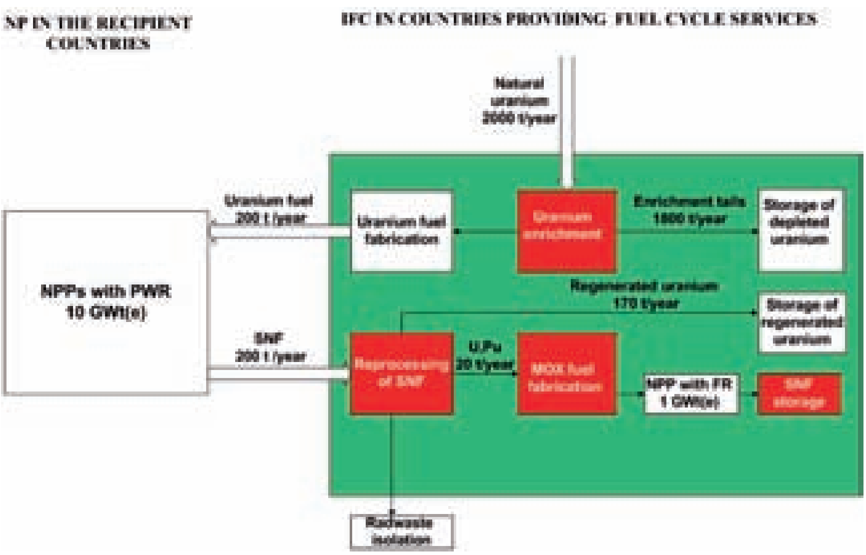

FIG. 2. Principal structure and balance flows in the nuclear energy system for a model international fuel centre providing uranium enrichment services and utilizing spent fuel from thermal reactors for energy production.

In addition to the advantages mentioned above, the scheme could substantially increase the cost effectiveness as well as the attractiveness of electricity generation by nuclear power in the eyes of the world community.

4. CONCLUSIONS

Interest in nuclear power as a key element of world energy supply appears to be increasing again. However, large scale use of nuclear energy in the 21st century will be restrained not only by the system and technological limitations inherent in the current approach to nuclear power, but also by political, infrastructure and economic restrictions.

The possibility of coping with these limitations and providing comprehensive solutions to the problem of energy supply for globally sustainable development can be achieved by the creation of a Global Nuclear Infrastructure, including the organization of international fuel centres providing uranium enrichment services as well as the generation of electricity through the use of spent fuel under IAEA safeguards.

The creation of a large scale worldwide nuclear infrastructure for sustainable development would be impossible without extensive international cooperation. The initiative of the President of the Russian Federation in conjunction with the President of the USA's Global Nuclear Energy Partnership (GNEP) could form, with significant IAEA participation, the political foundation for this cooperation. The result could be an optimized framework for the global future use of nuclear power.

DISCUSSION

J. BOUCHARD (France — Chairperson): Regarding international fuel cycle centres, does 'international' mean that they will be managed internationally or that they will be open to international use? How do you see the management of such centres?

A. ZRODNIKOV (Russian Federation): There are several possibilities, which are open for discussion.

A. GONZÁLEZ (Argentina): Like Mr. Brown before him, Mr. Zrodnikov made no mention of an international nuclear safety regime. What we have at the moment is just an embryo, but without a solid safety regime I do not see how the Russian President's proposal can succeed.

A. ZRODNIKOV (Russian Federation): I see an important task for the IAEA there.

L. VAN DEN DURPEL (Belgium): In the presentations of Mr. Pellaud and Mr. Brown, fuel leasing meant that the spent fuel was taken back by the manufacturing country. In the IAEA multilateral approaches document, however, there is also talk about a 'take-away' — as opposed to a 'take back' — approach, whereby the spent fuel would be sent by the leasing country to a third country. At this conference, there has so far been no mention of the 'take away' approach, for which the institutional arrangements would be different from those for the 'take back' approach. Why is that?

L. BROWN (USA): Some of the international legal difficulties in returning spent fuel or moving it across borders have already been mentioned this morning. The GNEP initiative is going to take a decade or two to produce the technologies that would make fuel leasing with take back viable from our perspective, so obviously we would also need to work in the international agreements area at the bilateral or the IAEA level. But, as I understood it from Mr. Zrodnikov's presentation, the Russian Federation is, or soon will be, able to take back spent fuel under arrangements made bilaterally or multilaterally. I think that might be a model that the rest of us could work towards.

J. BOUCHARD (France — Chairperson): This kind of 'circulation' has already been done in the past, with the fuel provided by one country being used in another country and then sent to a third country for reprocessing. I did not hear in the presentations of Mr. Brown and Mr. Zrodnikov anything that would seem to preclude that possibility, which should be kept open even if it would involve difficulties, because we will need such flexibility in the future.

R. EINZIGER (USA): With this particular system there is going to be a lot more transport of fuel. What about the resulting additional security risks?

L. BROWN (USA): Obviously, if you do any more than what you are doing now, it is going to involve new challenges — transport being a significant one.

BACK END OF THE NUCLEAR FUEL CYCLE: THE FRENCH CHOICES

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Abstract

The paper summarizes the strategies adopted in France with regard to the management of the back end of the nuclear fuel cycle. It gives an overview of French energy policy and indicates the merits of its emphasis on nuclear power. It describes the reprocessing–recycling approach being used at present, its technical and economic benefits, and the plans for managing the nuclear fuel cycle in the future. The arrangements in France for radioactive waste disposal are described, including the implications of the recent Act on the subject. Finally, the objectives and principles for managing the back end of the nuclear fuel cycle in France are set out.

1. BASIC PRINCIPLES OF THE FRENCH ENERGY POLICY

At the start of the 1970s, against a backdrop of a major oil crisis and a shortage of energy resources on French territory, France developed an ambitious energy policy. The following three objectives lie behind this policy:

(1) To increase the nation's energy independence and ensure security of supplies;
(2) To provide energy at competitive prices both for industrial and private use;
(3) To protect the environment.

These three objectives guided the major strategic decisions made at this time, i.e. the implementation of a significant civil nuclear programme for electricity generation purposes, the launch of an action plan to manage energy demand and, last but not least, the development of a hydroelectric potential.

These decisions had major consequences. The French energy bill amounted to €38.3 billion in 2006, i.e. 2.26% of the gross domestic product, compared to 5% in 1981. The energy independence level reached 50% in 2005 compared with 26% in 1973. The electricity generated is competitively priced and is not very subject to fluctuation as a result of changes in the price of oil. It covers all national requirements and France is the world's leading exporter of electricity. Finally, although France is now the 4th ranked energy consumer of all the OECD/NEA countries, it is only 27th when it comes to emissions of CO_2 compared with gross domestic product.

This energy policy is primarily based on the development of a substantial nuclear sector comprising 58 power reactors representing an installed capacity of 63 GW(e), which generated 451.5 TW·h in 2005, or in other words 78.5% of total electricity production. France also has a fast neutron reactor, Phénix, with a capacity of 250 MW(e), which is currently used for research into advanced partitioning/transmutation technologies.

Finally, apart from these industrial developments a legislative framework and an administrative organization structure have been established in order to satisfy the two major issues associated with the use of nuclear power, namely the existence of a high standard of safety, subject to continuous improvement, and a national policy for managing the back end of the nuclear fuel cycle. In the remainder of this paper the last subject is examined in greater detail.

2. THE REPROCESSING–RECYCLING OPTION AS A SOLUTION TO SPENT NUCLEAR FUEL MANAGEMENT

Alongside the development of its civil nuclear programme, France has opted to reprocess spent fuel. At present, this operation is carried out at a large industrial facility based at La Hague in western France. Two plants known as UP2 800 and UP3 are currently being operated on the La Hague site by the French AREVA group, with an annual capacity of 1700 t of spent fuel. Since 1976, 21 600 t of spent fuel have been reprocessed in these plants, which in energy terms corresponds to the equivalent of four years of oil production in Kuwait. This industrial facility has also led to the formation of a major technological and economic area and is currently directly responsible for more than 8000 jobs. Reprocessing operations have been performed on behalf of the French electricity operator, EDF (11 700 t), but also for foreign clients, and countries such as Belgium, Germany, Japan, the Netherlands, Sweden and Switzerland (9900 t) have concluded reprocessing contracts with AREVA since the mid-1970s.

2.1. The principle behind spent fuel reprocessing

After irradiation in a power reactor, spent fuel still contains 97% reusable materials which are capable of providing energy (the spent fuels from pressurized water reactors contain 96% uranium and 1% plutonium). The reprocessing procedure used in France is the PUREX process, and this entails separating these materials from the final waste, which represents the remaining 3% and consists of minor actinides and fission products. This waste is currently packaged in strong vitrified matrices which are specially adapted for long term waste management.

The uranium contained in the spent fuel contains slightly more U-235 (1%) than natural uranium (0.7% U-235); it is therefore still valuable in energy terms. Direct recycling of the reprocessed uranium (URT) is technically possible, and reloads of fuel assemblies manufactured using URT have already been successfully recycled in French reactors. For foreign clients of AREVA, much of the uranium requiring reprocessing has also been recycled by these clients in their reactors (4750 t out of 7350 t of uranium processed), and the rest will be recycled over the coming years. This is, therefore, a tried and mature technology.

The plutonium also has a high energy value (just one gram of plutonium can produce as much electricity as a tonne of oil) and it can also be recycled on an industrial scale via the use of mixed oxide fuel (MOX), either in the 20 French pressurized water reactors operated by EDF which use 30% MOX, or in foreign countries which have opted for the reprocessing–recycling option, e.g. Belgium, Germany and Switzerland. Since the La Hague reprocessing plant started operation, 118 t of plutonium have already been recycled in this form. A total of 35 European nuclear power plants have been authorized by their national safety authorities to use MOX type fuel. AREVA has a new plant, MELOX, in the south of France, which is responsible for manufacturing this MOX fuel. In summary, France now has a coherent, industrially mature and competitive set-up for its reprocessing and recycling technologies.

2.2. Benefits of spent fuel reprocessing

2.2.1. Rational management of natural uranium resources

The uranium and plutonium present in 1 t of spent fuel have the same energy value as 20 000 t of oil. The reprocessing–recycling strategy therefore offers a real potential for solving the problem of resources. It means that savings of approximately 20% can be made on the natural uranium resources needed to operate the French nuclear reactor network.

While this aspect may have seemed of secondary importance or outdated only a few years ago in a market where prices were very low, recent changes in the price of uranium confirm the benefits of reprocessing for the rational management of resources. The price of uranium increased from US $6 to

US $45 per pound of U_3O_8 at the time of the first oil crisis and dropped back down to US $7 per pound of U_3O_8 in December 2000, but was at around US $41 per pound at the end of April 2006. Furthermore, faced with the prospects of a developing nuclear power sector — in countries such as China or India, for example — and the need to control emissions of greenhouse gases, the reprocessing–recycling option provides an answer in advance of the development of the new generation of fast neutron reactors.

2.2.2. Optimized management of the back end of the fuel cycle

The reprocessing option has also allowed the optimization of the back end of the nuclear fuel cycle. Spent fuels are collected on a centralized site — the La Hague site — prior to being reprocessed. This means that the amounts of spent fuel present on the 20 nuclear sites currently operated by EDF are limited, and this facilitates the management and monitoring operations on these sites.

Reprocessing also results in a reduction of the space required for interim storage of the unloaded fuel. The plutonium from seven uranium oxide fuel assemblies is required in order to produce one MOX fuel assembly. A small amount of spent fuel is therefore still placed in interim storage. The policy of in-line plutonium recycling via the manufacture of MOX fuel and ongoing developments with regard to burnup rates leads to a near balance in the levels of spent fuel which are currently in interim storage at La Hague awaiting reprocessing.

2.2.3. Economic management of the back end of the fuel cycle

A great deal of international research work has been done to examine the costs associated with the back end of the fuel cycle, especially in comparing the reprocessing–recycling solution with the open cycle strategy, which involves direct disposal of the spent fuel. The majority of these studies conclude that there is a slight difference between the cost of electricity produced by nuclear means in an open or closed cycle. This is due to two terms with opposite effects when calculating the cost of electricity. In the case of direct disposal there are no reprocessing costs, but the fuel requires longer interim storage for cooling purposes and the disposal facilities are more expensive. To be more specific, an economic assessment conducted in France in 2003 on reference costs for electricity production showed that, in the case of a series of 10 third generation reactors of the European pressurized water reactor (EPR) type, assuming a discount rate of 8%, the cost per kilowatt hour of nuclear electricity is €28.4/MW·h, with fuel cycle costs representing €5.1/MW·h, or 18% of this figure, and reprocessing accounting for €0.9/MW·h, or approximately 3% of the reference cost. In summary, the recycling–reprocessing operations result in savings of 20% to 30% on raw materials set against an increase in production costs of only 3%.

2.2.4. Safe and sustainable management of the resulting final waste

Finally, reprocessing has led to waste packaging methods that are both compact and effective in the long term. For the French nuclear reactor network as a whole, and assuming that these facilities will have an average operating life of 40 years, a maximum of approximately 80 000 m^3 of intermediate level long lived waste will be produced, along with 6300 m^3 of high level vitrified waste. Compared with an open cycle, the volume of final waste is reduced by a factor of 4 and the radiotoxicity of this waste is reduced by a factor of 10. In addition, the surface area required by a disposal facility in a geological formation deep underground is also reduced (by a factor of up to 2 or 3 in the latter case). Finally, vitrified waste represents a robust and resilient waste package suitable for long term disposal: research conducted on this subject by the Commissariat à l'énergie atomique (CEA) has suggested that this glass matrix will retain 99.9% of its integrity after a period of 10 000 years.

2.3. Non-proliferation

The decisions made in France have led to a reprocessing process that complies with international standards on non-proliferation. Recycling arrangements (via the conversion of 20 French nuclear reactors to MOX) have been developed to permit in-line management of the plutonium resulting from reprocessing operations, thereby avoiding the formation of separate stocks of plutonium. Furthermore, when irradiating the MOX fuel making up around 30% of the reactor core, approximately 30% of the plutonium is used up in the operation, representing a positive balance from a non-proliferation point of view, as well as from an economic viewpoint, as this makes it possible to generate approximately 10% of electricity from nuclear origins. Finally, the PUREX industrial process is already suitable for adaptation to an evolutionary third generation formula, known as COEX, which avoids any need to separate the plutonium.

2.4. The future of reprocessing: Third and fourth generation processes

To conclude, opting for the reprocessing–recycling option has placed France in a particularly flexible position at the present time.

In the event that future energy policy decisions lead to the use of fast neutron reactors over the coming decades, spent fuel which is currently in interim storage (and particularly spent MOX fuel) would provide the necessary strategic energy materials to allow the operation of these reactors.

In addition, apart from the trends which have already been identified towards third generation processes, the PUREX process also provides a foundation for developing fourth generation processes compatible with a strategy which includes recycling and transmutation of minor actinides.

3. IMPLEMENTATION OF A NATIONAL RADIOACTIVE WASTE MANAGEMENT POLICY

In addition to developing a high performance industrial management system for the back end of the nuclear fuel cycle, France also intends to implement safe and controlled management of the various types of radioactive waste generated by nuclear activities. This has been achieved primarily by setting up a public body, the Agency for Radioactive Waste Management (ANDRA), with responsibility for such issues, and by passing the law of 30 December 1991 concerning long lived high and intermediate level waste.

3.1. Appropriate management for each type of radioactive waste

In order to manage its radioactive waste properly France has opted to develop a classification system based on two characteristic parameters:

(1) The activity level, which gives an indication of the intensity of the ionizing radiation at a given time and thus its potential ‘hazard’ rating. Four levels have been defined: very low level, low level, intermediate level and high level radioactivity.

(2) Half-life, which can be used to deduce the length of time for which the waste may remain hazardous. This is because the radioactivity of materials decreases over time. Three levels are used: very short lived for waste with radioactivity levels which halve in 100 days or less, short lived for waste with radioactivity levels which halve in less than thirty years and long lived for waste with radioactivity levels which halve in thirty years or more.

ANDRA has been surveying the radioactive waste in France for the past ten years or so. This has led to the publication of national reference inventories which provide a detailed overview, by category and by producer, of the existing quantities of radioactive waste and their locations.

This classification and related work on inventories make it possible to develop appropriate management solutions for each category of radioactive waste. Very low level waste and low and intermediate level short lived waste are managed in surface waste repositories located in eastern France. These account for 84% of the volumes produced. As for the remaining 16%, the aim is to provide disposal facilities just below the surface for long lived low level waste (with industrial commissioning anticipated for 2013), and to implement the solutions covered as part of the 1991 Act for long lived high level waste (with particular emphasis on deep geological disposal for final waste after a preliminary period of cooling via interim storage). While awaiting these long term management facilities, these types of waste are currently placed in interim storage at several sites, including Marcoule and La Hague, located in the south and west of France respectively. These interim storage facilities have a planned operating life of at least 50 years, during which time it is expected that long term management solutions will be defined.

3.2. A specific Act for long lived high and intermediate level waste

3.2.1. Three research routes

The management of long lived intermediate and high level waste (with characteristics of a high level of radioactivity and long half-life) is specified in the Act of 30 December 1991. This act defined a major and diverse research programme based on three different routes:

(1) Route 1 examines advanced separation/transmutation: this entails examining the possibility of reducing the hazard posed by waste by separating the most hazardous and long lived elements with a view to transforming these into radioactive elements with a shorter half-life by irradiation in reactors;
(2) Route 2 examines deep disposal;
(3) Route 3 examines waste packaging and long term surface interim storage.

Research into routes 1 and 3 is coordinated by the CEA, while route 2 is coordinated by ANDRA. More than €2.5 billion was spent on these research programmes between 1992 and 2004, including €810 million on route 1, €1007 million on route 2 and €672 million on route 3.

This research was based on initial experiments. The Marcoule site was used to conduct irradiation experiments for transmutation tests using the

Phenix experimental fast neutron reactor, and an innovative laboratory (ATALANTE) was constructed on the same site by the CEA to study advanced separation processes. As part of research into disposal techniques an underground laboratory was constructed in eastern France on the border between the départements of Meuse and Haute-Marne; this is located 500 metres deep in clay and provides an initial experimental facility for conducting scientific collaboration projects, including projects which will be developed over the course of the next 20 years or so, some involving international partnerships.

3.2.2. Results obtained

The Act of 30 December 1991 stipulated that a parliamentary debate should be held no later than 2006 in order to provide an update on the results obtained and to approve the implementation of the management solutions for long lived high level waste, where applicable. In June 2005, the organizations behind this research, the CEA and ANDRA, submitted overview reports of the research conducted over the past 15 years to the French Ministers for Energy and Research. These publications are available on the web sites of the two organizations in question (www.andra.fr and www.cea.fr) and have led to various independent assessments, notably by a special commission which was established as a result of the 1991 Act, the National Evaluation Commission, by the Nuclear Safety Authority and finally by international experts commissioned by the OECD/NEA. All of these assessments are available at the following web site: www.loi-dechets-radioactifs.industrie.gouv.fr.

On the subject of industrial interim storage and packaging processes, the main results arising from this research and confirmed by the assessments are as follows:

- Research into packaging has, amongst other things, made it possible to reduce the annual volume of long lived high and intermediate level waste by a factor of more than six, compared with initial expectations when the La Hague plants were originally designed.
- It has also been possible to establish the very long term behaviour of the vitreous matrix for high level waste. The studies indicate that these packages will take several hundred thousand years to dissolve under the conditions applicable to deep disposal in a clay environment.
- The operating life of the most recent interim storage facilities has been assessed as a hundred years or so.

As far as long term interim storage is concerned, research has been conducted into concepts with long operating lives (extending up to 150–300 years) and in technical terms such facilities could be constructed fairly rapidly and be operational within ten years or so once the decision to proceed has been taken. Nevertheless, at the end of this 150 to 300 year period the packages would still need to be recovered, a prospect which is not without its own problems in view of the social acceptability of this kind of management solution.

As for research into deep geological disposal, it has been demonstrated that the clay formations studied in Meuse/Haute-Marne, already 150 million years old, have properties which make them conducive to potential underground disposal — a homogeneous formation, particularly low seismic risk, suitability for excavation and low permeability. Finally, design studies have made it possible to define simple but strong repository structures which are suited to the clay formation. On this basis, ANDRA considered that such a disposal facility could be regarded as being feasible, and this was subsequently confirmed by the various assessment bodies in 2006. At the request of the French public authorities the concepts developed by ANDRA are reversible. The aim is to enable the waste to be recovered at a later date in the event of some unforeseen circumstance or if new solutions were to be developed as a result of progress in the field of radioactive waste management. This reversibility means that it is possible to move forward cautiously, strengthening, in particular, knowledge acquired during the research phase through experience and observation. Such a reversibility phase could be maintained for two to three centuries without the need for any major work on the facilities.

Finally, research conducted into advanced partitioning and transmutation has enabled the development and testing of molecules, allowing selective extraction of the various minor actinides. This research has also established the extent to which these elements might be ‘transmutable’: an operation of this kind would be conceivable for some of these elements (minor actinides) but seems unlikely for others (fission products). Further research work is still required in order to achieve success in the various areas. It seems unlikely that any industrial facilities will be commissioned much before 2040 (advanced processing plant and fourth generation fast reactors capable of transmuting minor actinides).

3.3. A new Act currently passing through Parliament

As required by the 1991 Act, France organized a parliamentary briefing during the first half of 2006 to provide updates on the research results obtained with a view to drawing up a new Act concerning the plan for the next 20 to

30 years. A bill was therefore submitted by the Minister for Industry, François Loos, on 22 March and then voted on by the French Parliament on 15 June 2006. It was drafted on the basis of the following elements:

(a) The reports submitted by the research organizations in 2005;
(b) The reports of the independent assessors following their scrutiny of the publications;
(c) The conclusions of a public debate which was arranged during the last quarter of 2005 to provide information on these issues to French citizens and allow them to express their concerns.

The main themes of this new Act are described below.

3.3.1. A national plan for radioactive waste and materials management

The Act establishes the principle of a national plan for radioactive waste and materials management to be drawn up by the French Government every three years, submitted to Parliament and transcribed in a regulatory text to permit its outline conclusions to be implemented. The fundamental principles on which this plan will be based are as follows:

— In the first instance, in order to reduce the quantity and hazardous nature of the waste, spent nuclear fuel from power plants must be reprocessed before it can be recycled in power plants;
— Secondly, waste which cannot be recycled must be packaged in strong and stable matrices and placed in interim surface storage;
— Thirdly, after interim storage, those parts of the final waste which cannot be stored permanently in surface repositories or just below the ground will be placed in deep geological disposal facilities where they must, initially, be retrievable.

Another major principle underlying the plan is that the ban on storing foreign waste in France is confirmed in the bill and legislation on this subject is strengthened. It stipulates that reprocessing of spent fuel from foreign countries will be subject to intergovernmental agreements which will specify limits for interim storage of these materials and the resulting waste after reprocessing. These limits will be specified on an individual basis as a function of the technical constraints associated with the reprocessing and transporting of these substances. The bill sets up a system of controls and sanctions which was not in the 1991 Act.

3.3.2. A research and work programme

The bill foresees research along three different routes, depending on their respective levels of maturity. Different landmarks are defined for achieving the specific targets of the research work:

— 2012: A full technical and economic overview of advanced partitioning and transmutation, and a decision on the relevant technologies (partitioning process, transmutation method by homogeneous or heterogeneous means, in fast reactors or hybrid systems);
— 2015: Modification or creation of new interim storage facilities;
— 2015: An investigation into an application for a construction licence for a deep geological disposal facility;
— 2020: Commissioning of demonstration facilities for advanced partitioning and transmutation processes;
— 2025: A waste repository coming into operation.

3.3.3. Independent evaluation of research, public information and consultation

The bill maintains and reinforces the principle of independent scientific evaluation as already stipulated in the 1991 Act and means of providing local information. The bill also specifies a detailed approval procedure for disposal projects. It stipulates that applications for construction licenses will be subject to public debate, two technical evaluations by the National Evaluation Commission and the safety authority, approval by the competent local and regional authorities, a law specifying the retrievability conditions for the repository and finally a regulatory text authorizing the facility. Disposal must be retrievable for a period of no less than 100 years and the facility may only be closed by an Act of Parliament. Such a system means that the French Parliament will remain closely involved with this long term procedure and also ensures that the competent local or regional authorities will have a part to play in the decision making process.

3.3.4. Financial aspects

The resources required to conduct the research work coordinated by ANDRA will be obtained by means of taxes levied on basic nuclear installations and paid into a fund for use by ANDRA. The bill also includes a means of safeguarding the finances for dismantling costs and industrial waste management expenses. Given the amounts at stake, over €30 billion set aside in the accounts of EDF, AREVA and the CEA according to a report by the

French Audit Office, and the distant dates on which certain expenses are likely to arise, it is essential that these sums be safeguarded. The cost of disposal itself is estimated to be approximately €15 billion in gross terms and €4 billion as a discounted value. The French industrialists in the nuclear sector must therefore not only make periodic and cautious assessments of their total nuclear expenses, and make appropriate provisions, but must also have financial assets to cover the full amount of these provisions.

These assets will be assigned exclusively to covering dismantling and waste management costs, which means that they may not be used by operators for any other purpose and may not be claimed by creditors, irrespective of the nature of their claims. These assets must also have adequate security and be sufficiently diverse and liquid. These measures will be monitored by the French public authorities.

4. MAIN PRINCIPLES FOR THE BACK END OF THE FUEL CYCLE

Given the depletion of fossil fuel resources and the pressing need to combat the greenhouse effect, all against a backdrop of growing demand for energy, there is a resurgence of interest in nuclear power in the world. Against this background, France supports the various international initiatives which facilitate the easy access to nuclear power by interested countries. Moreover, on the basis of its experience in the different aspects of nuclear power (reactor construction, front and back ends of the fuel cycle) and in the necessary regulatory framework to ensure that this energy source develops in a safe and controlled manner, France hopes to provide concrete proposals in this field.

4.1. A major objective: Nuclear safety

The various international initiatives currently under discussion address particular attention, quite legitimately, to the question of non-proliferation. However, the absolute first priority should not be forgotten: nuclear safety. This is why many bilateral agreements have been concluded between France or its agencies and their foreign counterparts. Amongst other things, these agreements aim to allow States developing a nuclear programme or setting up projects along these lines to take advantage of French experience with regard to governance in this sector. The French government, public research organizations (and particularly the CEA, the Nuclear Safety Authority and its technical support agency the Institute for Radiation Protection and Nuclear Safety) and the industrial operators (AREVA, EDF) have established many contacts and collaborate with many different countries. This collaboration represents a major contribution to developing a nuclear energy system that complies with the necessary safety and security conditions.

4.2. A responsible policy for the back end of the fuel cycle

Public opinion is extremely sensitive to the question of radioactive waste. Like France, the majority of nuclear States have adopted laws prohibiting the disposal of foreign radioactive waste on their territories. This principle has just been reaffirmed and reinforced in France as part of the Act on sustainable management of radioactive materials and waste, which was adopted by the French Parliament on 15 June 2006; this is fundamental to the acceptance by the French public of the use of sustainable management solutions for radioactive waste, including deep geological disposal, and eventually of the use of nuclear energy.

4.3. Regional disposal sites

Not all countries will have access to disposal sites, either due to the geological characteristics of their territories or due to the scale of their nuclear programmes. The creation of regional disposal sites restricted to certain countries wishing to pool their resources may represent one solution which complies with optimum safety and security conditions and avoids passing on the responsibility of inadequate waste management to future generations. Civil society is unable to accept the fact that certain States may discharge this responsibility by managing or entrusting responsibility for managing this waste under conditions that do not guarantee protection of the environment. It is essential that the ‘polluter pays’ principle be applied religiously in this particular field. In addition, restricting such waste repositories to a regional scale would make it possible to restrict the transport of nuclear materials and waste, which are sensitive operations, not only from the point of view of safety, but also of security and non-proliferation.

This leads to the following:

(a) There is a need to link the use of these regional waste repositories to the development of shared safety and security standards. To achieve this, the standards drawn up by the IAEA could form a foundation for a harmonization and ongoing development exercise, if necessary, in line with the best existing practices, based on the model for possible developments on a European scale, as demonstrated by the WENRA regulators’ initiative. A research programme could thus be launched along these lines.

(b) There is a need for States using these international waste repositories to behave responsibly in accepting all consequences arising from their use of nuclear power. This should take the form of a mandatory contribution on the part of these States to the necessary research and development costs, along with construction and operating costs. Industrial partnerships between the various countries involved in a regional disposal scheme would thus be formed and these might possibly take the form of industrial and financial participation schemes linking the countries in question to the company operating the repository.

(c) There is a need to consider the appropriate regional scales with a view to minimizing necessary transport operations.

4.4. Partnerships in the back end of the fuel cycle

France is equipped with a mature and efficient industrial tool that covers the whole fuel cycle. Regarding the back end of the fuel cycle, industrial partnerships were implemented in the past with European countries, as well as with Japan. The currently available reprocessing capacities at La Hague allow us to re-conduct such partnerships.

4.5. New technologies

The schemes described above may evolve either in the medium term as a result of possible adaptations of the current reprocessing processes or in the longer term with the use of advanced separation/transmutation processes. These permit the separation of minor actinides and their transmutation in fast neutron reactors. France has set itself ambitious targets in this respect, leading to technological decisions by 2012 (reactor types and separation processes), demonstration stages from 2020 and the prospect of industrialization by 2040. Finally, France is an active participant in international discussion groups on this subject.

DISCUSSION

J. BOUCHARD (France — Chairperson): You mentioned the fact that in the new French law there is a clear statement about the impossibility of France managing or disposing of foreign waste. What was the reason for this provision? It is not new — it was already in the previous law.

F. FOUQUET (France): I think that in France, and maybe in other countries, public opinion is very sensitive to the question of radioactive waste. When we organized the national public debate in the last quarter of 2005 there was a clear conclusion; people said “We are able to accept the construction of a deep geological disposal facility in the eastern part of France, but the condition is only to have in our facility French radioactive waste — we do not want to have other waste — each country has to responsible for its own radioactive waste.” We understood that to be a clear condition for French citizens, especially those living near the possible location of an underground disposal facility. That is why we decided to reinforce the provisions in the 1991 Waste Act, which were not clear enough and gave rise to a lot of disagreement.

J. BOUCHARD (France — Chairperson): Is the final disposal of radioactive waste the only point on which France is forbidding foreign contracts? For instance, we have had contracts with other countries in the past for reprocessing in France. Are some contracts still being fulfilled?

F. FOUQUET (France): The only issue is about the final disposal of waste. For spent fuel, reprocessing contracts with foreign customers are still possible.

H.G. FORSSTRÖM (IAEA): I am rather confused. I was about to ask whether France was going to participate in regional repositories, and then when I heard your answer to that question, I realized that it is not. But you made a number of conditions for the acceptance by France of regional repositories. What would be the role of France in such regional cooperation, or is it merely a sort of theoretical discussion about these conditions?

F. FOUQUET (France): I think it may be too early to speak about the role of France. I said that France had been involved in very important safety related bilateral cooperation and we are ready to strengthen our involvement through, for example, the provision of training. We are also ready to participate in safety studies relating to the safety of regional disposal facilities, and our industry could participate in the construction of such facilities. But France would not host a regional disposal facility.

A. GONZÁLEZ (Argentina): I am not as confused as Mr. Forsström — your presentation was the only one to clearly underline the importance of an international safety regime. I hope that the French law approved two days ago includes a provision relating to such a regime; the draft did not provide for it.

When you talked about the environmental benefits of reprocessing and recycling, you did not mention one of the main ones — much less uranium is needed for a given amount of energy produced, and the big environmental impact is from the mining and milling of uranium because of the public exposure to the tailings and because of the occupational exposure which occurs in uranium mining. That was recognized during INFCE about 20 years ago, but people have since tended to forget it.

S. SAEGUSA (Japan): Your national management plan defined three main principles — first, reducing waste quantity and toxicity, second, interim storage and third, disposal. When you say 'interim storage' do you mean the interim storage of spent fuel or of high level waste?

F. FOUQUET (France): We envisage interim storage both for spent fuel before reprocessing (so it is quite a short interim storage period) and for MOX fuel, which is not currently being reprocessed because we favour saving it for use in fast reactors. There is also a role for interim storage of high level waste because it will be necessary to wait 60 years to allow cooling before such waste is put it into a geological disposal facility.

That is why the second principle of the plan is safe interim storage both for spent fuel and for high level waste. But interim storage is only a temporary solution before geological disposal, which is the safe long term management solution, and that point was not clear in the past because the 1991 Waste Act put interim storage and deep geological disposal on the same level. What is new now is the decision that the safe long term solution is reversible deep geological disposal and before that interim storage for maybe 60 years.

J. BOUCHARD (France — Chairperson): This is a very important point. This morning we had a discussion about the 'wait and see' approach, but this is not a 'wait and see' approach. In France, interim storage is no longer considered to be a long term management option. It is only a form of flexibility for helping to cope with the real options, which are reprocessing with recycling and deep geological disposal.

EVOLVING POLICIES IN EXPANDING ECONOMIES

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Abstract

The rationale for the continuing expansion of the nuclear power industry in India is set out and the technical details of the progress to date are described. The paper sets out the national plans for increasing nuclear energy capacity in the coming years. This will involve embarking on a programme of fast breeder reactors and utilization of the thorium resources in the territory of India. The plans include continuing with the closed fuel cycle and reprocessing the spent fuel from thermal reactors. The strategy should result in a significant reduction of the amount of waste for which geological disposal is needed.

1. INTRODUCTION

The growing world demand for energy, coupled with dwindling supplies of fossil fuels and efforts to curb global warming, have rekindled interest in nuclear energy. In this context, the importance of nuclear safety, non-proliferation and security must always be stressed, especially in relation to the management of spent fuel. Such considerations must be taken into account by the energy planner so that effective and economical solutions that are globally acceptable can be developed.

2. INDIAN SCENARIO

India, with its population of 1.1 billion, is host to one sixth of the world's population. The population of India is expected to stabilize at around 1.5 billion by 2050. The country has seen impressive growth in the electricity sector since independence in 1947, the installed capacity growing from a meagre 1.7 GW(e) to around 130 GW(e), almost by 85 times. However, present per capita electricity generation in India is about 600 kW·h per year. As such, large capacity additions are required just to maintain the current level of sustenance. India's gross domestic product (GDP) has been growing at the rate of 6–10% since 1990 and the forecast is that it will continue to do so in the coming decades. GDP growth has led to a spurt in the consumption of primary energy as well as electricity. An addition of 10 GW(e) per year is planned in the coming years. This has necessitated developing a strategy for growth of electricity generation, based on a careful examination of all the issues related to sustainability, available energy resources, diversity of sources of energy supply and technologies, security of supplies, self-sufficiency, security of energy infrastructure, effect on local, regional and global environment, health externalities and demand side management.

While India produces about 2.5% of world energy, it consumes 3.3% of world energy. India is thus a net energy importer. Any attempt by India to even marginally increase the per capita generation of electricity through the fossil fuel route would involve the import of large quantities of fuel. The increasing use of fossil fuel would further aggravate environmental concerns.

To meet electricity needs, energy managers in India are adopting an optimal mix approach considering the uneven distribution of its natural resources, the economics of power generation and environmental concerns. Nuclear energy has been identified as being an important component of this mix to provide electricity generation at locations away from the coal mines, as a short term measure, and to serve as a major energy source in the long term. It is environmentally friendly, technologically proven, economically competitive and provides energy security. By and large, nuclear power has also gained social acceptance in this part of the world and there is no significant public resistance to the building of nuclear power plants (see Fig. 1).

With the envisaged GDP growth, a projected capacity addition of about 6.2 GW(e) by the year 2011–2012 has been planned. Based on past performance and the focused attention being paid to the electricity sector, that target is well within reach. Energy planners have envisaged increasing the share of nuclear energy from the present 3% to 20% in the next three decades.

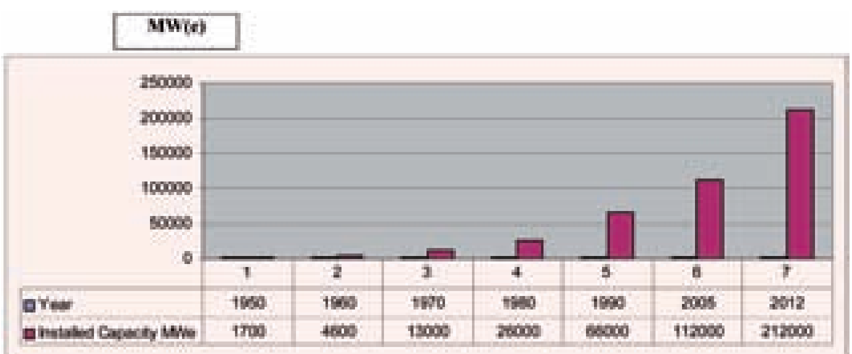

FIG. 1. Actual and planned capacity for electricity production in India.

3. POTENTIAL OF NUCLEAR ENERGY

Because it has modest reserves of uranium but vast reserves of thorium, India has embarked upon the closed fuel cycle. Accordingly, the Indian nuclear programme has been tailored to exploit uranium and thorium most efficiently through a three stage programme.

The first stage nuclear power programme is based on PHWRs using natural uranium as fuel, which can provide a capacity of 10 GW(e). It is being followed by a second stage programme based on fast breeder reactors (FBRs), using plutonium obtained through reprocessing of the spent fuel from the first stage. This will be followed by the third stage, based on uranium–thorium reactor systems to be established for utilizing the thorium resources. The entire cycle has the potential of supporting capacities of 300 GW(e) for 300 years.

4. DEVELOPMENT OF THE NUCLEAR POWER PROGRAMME

India’s nuclear power programme commenced with the building of two 200 MW(e) BWR units obtained on a turnkey basis from General Electric in 1964. In parallel, two 200 MW(e) PHWR units were built with the technical cooperation of Canada. Since then, the first stage programme has progressed slowly but steadily. The country has established comprehensive indigenous capabilities in design, fabrication of equipment, construction, commissioning, operation and maintenance of PHWRs. The country has developed its own design of PHWR with capacities of 220, 540 and 700 MW(e). At the present time, a nuclear power generation capacity of 3900 MW(e) is in operation with 16 units (two 160 MW(e) of the BWR type, one 100, one 200, ten 220, two Þ 540 MW(e) PHWRs), representing 265 reactor years of experience with no incident involving radiation release beyond internationally acceptable limits. Six units (four 220 MW(e) PHWRs and two 1000 MW(e) WWERs) at three sites are under construction. The two 1000 MW(e) units are being built with Russian cooperation. The two 540 MW(e) units at Tarapur have matched the international benchmark time of five years for construction. This experience has given impetus to further reduce the construction time to four and a half years for ongoing projects. The Government has recently approved the building of eight more units, comprising four 700 MW(e) PHWRs and four 1000 MW(e) units of LWRs, for which pre-projecta activities have started with the planned launching of the project at the end of this year or early next year.

Capabilities have also been developed in the renovation/modernization, upgrading and life management of ageing reactors. Major renovation and upgrading work has already been successfully completed on five reactors.

Also, in the area of the fuel cycle, an indigenous infrastructure has been developed for prospecting, mining and milling of uranium, fuel fabrication, production of heavy water, spent fuel reprocessing, radioactive waste management, and manufacture of control and instrumentation systems and equipment. Critical nuclear components are fabricated to stringent standards by the Indian industries.

Parallel to this development in the first stage programme of PHWRs, activities on the commercial launching of the second stage programme involving the building of the FBRs have begun. A fast breeder test reactor (FBTR) at Kalpakkam, with a capacity of 40 MW(th) (13 MW(e)) was commissioned, in 1985 and the unit is still in operation. Construction work on the 500 MW(e) prototype fast breeder reactor designed indigenously has commenced, and the reactor is expected to go on stream by 2010. This PFBR will be fuelled by recycled plutonium and depleted uranium recovered from the spent fuel of PHWRs. Initially, it is proposed to use plutonium oxide mix as fuel, and efforts are under way to develop metal fuel for use at a later stage to achieve a better doubling time. It is proposed to commission four more 500 MW(e) FBR units by 2020.

Design and development work on the advanced heavy water reactor (AHWR) is in progress for demonstration of the technology to utilize thorium for electricity generation. The AHWR will be a forerunner of the reactors to be set up in the third stage of the nuclear power programme.

The capacity potential of the first stage, based on indigenous uranium resources, is estimated to be about 10 GW(e). The deployment of the subsequent stages will be sequential with transition phases between them to develop and demonstrate the technology of each stage. Currently India is in the transition stage from the first stage (PHWR thermal reactors) to the second stage (FBR). During the transition period, to meet the large energy demands, it has been decided to import LWRs, and the building of two 1000 MW(e) WWERs with the cooperation of the Russian Federation is the first such venture.

5. PROPOSED NUCLEAR POWER PROGRAMME

India is pursuing the implementation of its nuclear power programme with a view to increasing the country’s nuclear power capacity from the present 3% to 20% by 2020. A nuclear power programme reaching a total nuclear capacity of about 10 GW(e) by the end of 2012 and about 20 GW(e) by the year 2020 has been envisaged, comprising 10 GW(e) produced by PHWRs, 2 GW(e) by FBRs and 8 GW(e) by LWRs. The target has recently been upgraded to 40 GW(e), with 20 GW(e) from LWRs expected through the import of foreign reactors.

6. MANAGEMENT OF SPENT FUEL

The management of spent fuel has become the single most important issue linked to the acceptability of the future growth of nuclear power. Currently, nuclear power is being considered as a serious contender as an alternative to non-fossil power sources to combat global warming. However, the public is not convinced or satisfied with the various technological options being discussed. The management of spent fuel is a very complex issue involving many technological, socioeconomic and environmental challenges associated with:

(a) Maintaining safety and security, which requires enhancement of national measures;
(b) Protection against potential hazards — not only for the present but also for future generations;
(c) Preventing radiological accidents with well engineered mitigating provisions.

The options being considered for the back end of the nuclear fuel cycle are:

(1) Closed cycle — in which the spent fuel is reprocessed and used as fuel for subsequent stages, thereby reducing the volume of HLW considerably. HLW is immobilized in glass and disposed of in deep repositories after being encapsulated in multilayer metal canisters.
(2) Open cycle — in which the spent fuel is required to be stored for a very long time until it can be disposed of in deep geological repositories.

Many countries at present have opted for an open cycle, but, considering the long term perspective with regard to the increase in uranium utilization, the waste management problems with spent fuel when considered as a waste and the economy of operations, the adoption of the closed fuel cycle option is considered to be inevitable. India chose the closed cycle option from the beginning, in view of the availability of large quantities of thorium on its territory.

7. INDIAN STRATEGY FOR SPENT FUEL MANAGEMENT

In the Indian context, as the country has chosen the ‘closed fuel cycle’, spent fuel is not considered to be a waste but a valuable source of energy. It is planned to reprocess the spent fuel. The plutonium, along with depleted uranium recovered from the spent fuel, will be utilized as fuel for fast breeder reactors in the second stage of the nuclear power programme. The FBR can increase uranium utilization by about sixty times compared with what is possible through the use of PHWRs. The closed fuel cycle option is also highly beneficial from the waste management perspective, as it significantly reduces the amounts of high level long lived waste to be immobilized by vitrification. At present, the amount of spent fuel generated through 265 reactor-years of operation is modest, as the total installed nuclear power capacity is only about 4000 MW(e). The capacity is likely to double in the next 2–3 years. The spent fuel generated so far has been stored in the following ways:

(a) Spent fuel storage pools: These are water filled storage pools (bays) located at each reactor. The storage pools are designed for the storage of fuel bundles under water, and the existing storage capacity can accommodate the spent fuel discharge from 10–15 years of operation, together with the capacity to accommodate the discharge of one entire core.
(b) Away from reactor fuel (AFR) pool. The spent fuel from the co-located spent fuel pools is transferred to those facilities which are designed to store spent fuel bundles for an additional 10–15 years of operation.
(c) Dry cask storage: Spent fuel, after enough cooling, is stored in the dry cask storage facilities.

8. OPERATING EXPERIENCE IN WET AND DRY STORAGE OF SPENT FUEL

8.1. Wet storage

The spent fuel bundles discharged from the reactors have traditionally been stored in wet storage facilities attached to the reactors. The experience with all the storage bays has been excellent, with extremely good visibility for underwater operation. These bays have the capacity to store the spent fuel for a predetermined period (ranging from 10 to 15 years), beyond which they will be moved to the central storage facility located alongside the reprocessing units. Improvements in the design of these bays have been carried out as required by the changes in safety requirements. There has been progress from a basic underground tank lined with stainless steel in the first reactor with boreholes for monitoring the leakage, to the present design of double walled bays (tank in tank) with interspace for monitoring leakage. Features such as single failure-proof cranes are being installed to improve safety. The underground pools are equipped with extensive cooling, purification, chemistry control and ventilation systems. To supplement the storage capacity in the reactor pool, India has built two underwater storage facilities located AFR at Tarapur and at Rawatbhata.

8.2. Dry storage

India has also designed concrete casks for the dry storage of spent fuel as a temporary measure to provide storage capacity for fuel until the construction of the AFR is completed. The concrete casks were designed taking into account all the scenarios encountered in handling the casks on-site and the maximum loads that would be imposed on them during such handling. Necessary tests to demonstrate their adequacy were also done and clearance was obtained from the regulatory body for the use of these casks. A storage yard was also developed and built to store an estimated 600 t of spent fuel. The necessary conditions, such as maximum burnup, minimum cooling period in the reactor pool, etc., were considered in the design of the facility and the conditions were closely complied with in subsequent operations. The facility and the dry storage casks were made with provisions for normal radiological surveillance and safeguards by the IAEA. About 200 t of spent fuel have been put into 88 casks and stored in the facility since 1994. The casks have a design life of 40 years. In order to assess the status of bundles stored in this way, two casks were recently unloaded for examination. It was observed that there is no deterioration of any kind in the bundles or the casks after 12 years of storage

9. SPENT FUEL TRANSPORT

The reprocessing facilities are situated several hundred kilometres from the nuclear power plants. This necessitates the transport of spent fuel through the public domain. Transport of radioactive material in India is governed by the Atomic Energy Regulatory Board (AERB). The shipping flasks are designed and prepared in such a way that the radioactive material remains contained during the whole process of transport to prevent contamination, and remains shielded to avoid unacceptable radiation exposure to cargo handlers and the public. The spent fuel shipping flasks used for transport of spent fuel have been designed and tested to withstand normal and accident conditions in transport, as required by national/international regulations. Extensive simulations using computer codes have been carried out. Tests were conducted on models of the flasks to validate the simulation results to qualify the flask design. Complying with these regulations and tests also ensures that the radiation exposure of the population from normal incident-free transport and from potential transport accidents is well within regulatory limits. India has gained significant experience in the transport of spent fuel by handling a large number of shipments.

10. RECYCLING

India has opted for reprocessing as the strategy for the back end of the fuel cycle as part of its nuclear power programme. Industrial scale reprocessing plants exist to reprocess spent fuel for obtaining the necessary plutonium and uranium for use in the first 500 MW(e) prototype fast breeder reactor, which is under construction. India is planning to add to its reprocessing capacity to keep pace with the need for plutonium for its further fast breeder reactors. Reprocessing will be done strictly in relation to needs linked with the expansion of the fast breeder programme.

The small scale recycling plant commissioned in the early stages of the industry has already provided the necessary experience for developing mixed carbide fuel for the FBTR and MOX fuel for the BWRs and PHWRs. The unique indigenously developed plutonium-rich mixed carbide fuel used in the FBTR has performed extremely well up to a burnup of around 150 000 MW·d/t without a single fuel pin failure. MOX for BWR was developed as a substitute for enriched uranium fuel and 12 MOX fuel bundles have been successfully irradiated. As a part of the development of higher burnup fuel for PHWRs, 50 MOX fuel bundles have been successfully fabricated and used. Thorium bundles were also fabricated and used in PHWRs to gain the necessary experience for the third stage of the nuclear power programme. This has allowed sufficient data and experience relating to the fabrication of plutonium and thorium to be obtained, as well as providing experience of the nuclear physics aspects of the use of these materials in the reactors. India has also successfully utilized the depleted uranium obtained by reprocessing spent fuel from PHWRs.

Another important achievement is the successful reprocessing of plutonium carbide FBTR fuel discharged at a burnup of 100 000 MW·d/t and thorium bundles to separate U-233.

11. WASTE MANAGEMENT

Radioactive waste generated at nuclear power plants in various forms — solid, liquid or gaseous — is of the low level (LLW) and intermediate level waste (ILW) categories. Very little high level waste (HLW) or long lived waste (LLW) is generated at the nuclear power plant sites. All the waste is disposed of employing well established disposal techniques. For the long lived high level radioactive liquid waste (generated during reprocessing of spent nuclear fuel), the vitrification process is used. Initially, this high level liquid waste is stored in underground storage tanks with provisions for cooling, level monitoring and transfer. After the decay of short lived radionuclides, this HLW is converted into a solid form. Different matrices such as glass, synthetic mineral and ceramics have been developed for the immobilization of HLW. Glass of borosilicate type and other compositions has been selected for industrial scale immobilization in India. This vitrified waste has been found to have the desired characteristics, such as very high radiation resistance, excellent thermal stability and zero leachability. The vitrified waste is being stored in a specially designed solid waste storage facility for appropriate periods before ultimate disposal in a repository. With regard to the repository, research and development work is in progress. The amounts of high level waste to be managed are small. Adequate time is available to choose the best technological option for the repository based on developments around the world.

12. ACCELERATOR DRIVEN SYSTEMS AND WASTE MANAGEMENT

Present accelerator technology can provide suitable proton accelerators to drive new types of nuclear systems to destroy nuclear waste or to produce nuclear energy. While technical solutions like plutonium partitioning and recycling, followed by geological disposal of high level waste, seem to be viable, India is concerned with the ethics of an approach that imposes the responsibility for hazardous waste on future generations for hundreds of years. In order to obtain the maximum possible burnup of long lived radiotoxic waste, in addition to recycling plutonium, the plan is to burn the minor actinides that appear in LWR discharges, i.e. neptunium, americium and curium. Research and development work to perfect subcritical reactors driven by high power accelerators for the purpose of transmuting americium, in a pure form or in a mixture with other actinides, is under way. India is hopeful that these accelerator driven systems (ADSs) may result in waste containing radionuclides with shorter half-lives and allow for a reduction of the amount of high level waste to be sent to the geological repository by a factor of 100.

13. USE OF NUCLEAR WASTE IN GAMMA IRRADIATORS

Food preservation and radiation sterilization plants require gamma sources. Waste products, such as radiocaesium and other fission products, are being converted into special form materials for use in the irradiators in place of conventional cobalt. It is expected that this effort will help in finding a good use for the waste that would otherwise have to be disposed of.

14. CONCLUSIONS

India has pursued a well defined path of a three stage nuclear power programme to exploit its naturally available resources fully. This has necessitated the adoption of the closed fuel cycle option. The management of spent fuel is required for an interim period before it is used for reprocessing. Currently, there are about 3500 t of spent fuel in wet and dry storage facilities. Industrial scale reprocessing plants are being established to extract the plutonium required for the first fast breeder reactor being built at Kalpakkam. It is planned that the existing reprocessing capacity will be augmented to meet the demand for plutonium by the additional fast breeder reactors that will be constructed. As the installed capacity of fast breeder reactors increases, the demand for storage of spent fuel is expected to decrease rapidly. The highly radioactive waste from the reprocessing plants is vitrified at various locations before it is placed in a solid storage surveillance facility. There will be benefits in the form of reduced amounts of high level waste due to the closed fuel cycle option that has been adopted. It is hoped that work on the utilization of radiocaesium in irradiators and ADSs will be successful and that the ultimate solution will be found for the growing problem of radioactive waste management.

DISCUSSION

J. BOUCHARD (France — Chairperson): You mentioned that for 2020 your goal could be higher than it was before. Is it related to a recent development concerning another kind of energy or is it due to new developments in the nuclear system?

S.K. JAIN (India): Our plan was to have a nuclear capacity of 20 GW by 2020. We have raised the target to 40 GW, hoping that India will gain broad access to foreign technology and be able to import 20 light water reactors of 1000 MW capacity, each to be set up in the next 10–15 years.

C. PESCATORE (OECD/NEA): What kind of fast breeder reactor technology are you using? Sodium cooling? Also, how big is your regulatory organization?

S.K. JAIN (India): We have a small FBTR which is sodium cooled. It was fuelled with plutonium carbide, which was later replaced by MOX fuel. The 500 MW(e) reactor which we are setting up is sodium cooled and will use MOX fuel to start with. We have plans to gradually shift to metallic fuel at some stage.

We have a strong regulatory body which has close interactions with other national regulatory bodies.

D. LOUVAT (IAEA): As India is not yet a Party to the Joint Convention, we do not know much about its radioactive waste management strategy. I understood from your presentation that all the fuel from India's PWRs will be reprocessed into fast breeder fuel and all fuel from the fast breeders will be reprocessed for use in thorium fuelled reactors, but what would eventually be the strategy for the last spent fuel?

S.K. JAIN (India): The final waste from reprocessing will go to a deep geological repository. But as I mentioned, the parallel strategy that we are working on is to use some of the fission products in supporting our gamma irradiation programme for agriculture. In the longer term, we will be working to reduce the actinides in spent fuel by the use of high energy beams.

R. EINZIGER (USA): What is the maximum burnup of the fuel you transport?

S.K. JAIN (India): As far as the heavy water reactor is concerned, it uses natural uranium and the burnup is only 15 000 MW. The fuel is in excellent condition even after 40 years of storage.

With regard to the plutonium carbide fuel that we have used in our fast breeder, after 150 000 MW·d the fuel is still in very good condition.

SPENT FUEL: NUCLEAR SECURITY ISSUES

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Abstract

The paper gives an overview of the current global security threats facing the nuclear industry, with particular reference to spent fuel. It describes the types of nuclear facilities and activities involving radioactive materials that are at risk and the international measures established to counter the risks. In particular, it describes the activities of the International Atomic Energy Agency aimed at assisting its Member States in the nuclear security area.

1. FOUR THREATS OF NUCLEAR TERRORISM

Nowadays, the threat of nuclear terrorism cannot be excluded from our world. One year it was a London subway, the year before a Madrid railway; these attacks took a toll of many innocent lives and caused substantial damage to public infrastructure. The same applies to earlier terrorist acts in Indonesia and the Russian Federation. The international community is concerned that one of the next terrorists targets could be a nuclear one. The terrorists might acquire a nuclear weapon or nuclear material to produce an improvised nuclear explosive device (IND) or obtain other radioactive material to produce a radioactive dispersal device (RDD). If used in a city the consequences of an IND explosion would be devastating in terms of direct human loss of life, while in the case of an RDD used in a city the damage would not be in terms of human loss of life but rather in the form of psychological, sociological and economic impact, due to the possible long term health effects, the need for relocation of the population in the affected area and the possible need for decontamination of streets and buildings. Similar concerns relate to the risk of the sabotage of nuclear or other facilities or of radioactive material in transport.

2. SPECTRUM OF TARGETS

Probably more than 120 000 nuclear weapons [1] have been produced in the world during the past 60 years. Many were dismantled after the Cold War, but the number of existing weapons is estimated to be in the range 27 000 [2, 3] to 32 000 [4]. These weapons and related nuclear material are clearly outside the mandate and Statute of the IAEA and also, at the moment, they are not influenced by the Nuclear Non-proliferation Treaty (NPT). It therefore must be assumed that measures are being carried out in States possessing these weapons and materials to address nuclear security at the level commensurate with the risk.

There are 442 commercial nuclear power plants operating worldwide in 31 States, with an installed capacity of 370 GW [5]. Six are in a state of long term shutdown and a further 27 are under construction, with a potential capacity of 21 GW [6].

Of the 679 research reactors in the IAEA's database [7], 169 are decommissioned, 248 are in operation and 239 are shut down, waiting for further decisions. Highly enriched uranium (HEU) is still used in 61 research reactors and is present at an additional 10 shut down research reactors. As part of the initiatives to eliminate the use of HEU in research reactors, some States have managed to send their spent fuel and the HEU fuel back to the States of origin, the USA and the Russian Federation. The IAEA's database [8] shows that there are 8 operating reprocessing plants, 3 in a standby mode and 7 in shutdown mode.

The numbers of radioactive sources are much larger [9] and so, therefore, are the potential targets. In the IAEA Safety Standards the sources are divided into categories, depending on the associated activity, D. The number of category 1 sources (having activities above 1000 D) is estimated to be over 10 000 [11–13]. These include industrial and food processing irradiation facilities, medical teletherapy units and gamma knives [14], and radioisotope thermal generators (RTGs). The IAEA's directory of gamma processing facilities [15] shows 123 facilities in 45 Member States (ranging from 10 kCi to several MCi). The IAEA NAFA database [16] lists 33 Member States with 70 authorized food irradiation facilities. The number of RTGs produced in the Russian Federation [17] is estimated to be 1000. So far, more than 100 RTGs have been dismantled and transported to safe and secure storage [18]. The number of RTGs produced in other countries is not known, but published sources show that 44 RTGs were used in the USA's space programme (30–300 kCi) [19].

The number of category 2 sources (with activities between 10 and 1000 D) is estimated to be over 100 000. These include industrial radiography devices and high and medium dose rate brachytherapy units. Finally, the number of category 3 (with activities between 1 and 10 D) sources is estimated to be over 1 000 000. These include industrial gauges and well logging sources. In total, there are more than 3 million radioactive sources worldwide.

3. IS THERE A NEED TO BE CONCERNED FROM A NUCLEAR SECURITY PERSPECTIVE?

At the moment no commercial high level waste or spent fuel repository is in operation, although a few Members States have promising plans, have done research and have even made licensing decisions. The amount of spent fuel at nuclear power plant sites is steadily increasing, and at some reactor sites there will soon be the equivalent of more that 10 cores of spent fuel stored on-site with very large inventories of medium and long lived radionuclides. Long term storage approaches include increasingly densely packed storage at existing spent fuel pools, dry storage facilities or storage casks on-site or off-site. In addition, many designs have spent fuel pools located outside hardened containments.

Nuclear power plants have so far produced over 270 000 t of spent fuel containing 1800 t of plutonium [20]. About one third of the spent fuel has been reprocessed so far, separating several hundreds of tonnes of plutonium, mainly for MOX fuel. The risk posed by separated plutonium is a specific issue and will not be addressed here. It appears that for some time into the future, increasing amounts of spent fuel will be stored at the surface.

Several studies were done in Germany, Switzerland, the UK, the USA and other countries on the subject of threat assessment to nuclear power plants and spent fuel pools, including threats associated with commercial aircraft [21–25]. For obvious reasons, the details of the studies are not in the public domain; on the other hand it makes the exchange of information among those needing to operate and protect the facilities more difficult. The main risk to spent fuel is from sabotage to dense storage spent fuel pools. This could be from an attack on the site or from a stand-off attack using civilian aircraft. Some studies have explored worst case scenarios including damage or collapse of the pool, loss of coolant in the pool, overheating of the fuel, burning of Zr cladding, major release of Cs-137, Sr-90, Am-241 and other radionuclides from the spent fuel. The results show that the potential vulnerabilities and consequences are very much plant specific, but the conclusion is that spent fuel pools cannot be dismissed as valuable targets to terrorists. Several measures have been taken by States to respond to these new threats, but the fact remains that backfitting the existing pools with hardened engineering solutions is very difficult.

4. IS THERE A NEED TO WORRY IN THE FUTURE?

If there is going to be a nuclear power renaissance [26–29] in the light of climate change, growing population, growing living standards and shortage of fossil fuels, then by 2050 there may be between 1000 and 1500 nuclear power plants with an installed capacity in excess of 1000 GW [30] and 1 000 000 t of spent fuel accumulated, including 10 000 t of plutonium [20].· If such scenarios were to come into being there would be not only more power plants, but many more fuel cycle facilities, including plants for reprocessing civilian spent fuel, and there would be more fast breeder reactors to burn plutonium and the minor actinides. This would automatically mean much more transport of spent fuel, MOX fuel and high level waste. This would also mean the introduction of nuclear power facilities in several new countries that have yet to establish industrial and regulatory infrastructures with embedded safety and security cultures.

So the spectrum of targets that the potential adversaries could use for theft of nuclear or other radioactive material or other malicious acts, including sabotage to nuclear power plants, research reactors, spent fuel facilities, reprocessing facilities and transports, would be quite large.

5. INTERNATIONAL NUCLEAR SECURITY REGIME

Security is the responsibility of the State. However, in establishing an effective international system to combat the threat of nuclear terrorism, States need to work together. During the past few years a global nuclear security regime has emerged, with legally binding instruments such as the Convention on Physical Protection of Nuclear Material (CPPNM) [31], with 119 States Parties, the CPPNM amendment [32] adopted by consensus in 2005, the Nuclear Terrorism Convention [33] of 2005, with over 100 signatories, and UN Security Council Resolutions (UNSCR) 1373 and 1540. The Code of Conduct on the Safety and Security of Radioactive Sources represents an essential complement to these instruments as a political commitment of States to implement safety and security infrastructures and measures to effectively control radioactive sources. These instruments were developed primarily to address sub-State actors — terrorists or criminals — to prevent, detect and respond to malicious acts involving nuclear and radioactive material and facilities.

This framework is further enhanced by complementary safety instruments such as the Early Notification and Assistance Convention and the Convention on Nuclear Safety, and the Joint Convention on the Safety of Spent

Fuel Management and the Safety of Radioactive Waste Management. The latter two conventions are of particular importance because of the established periodic review mechanisms of the Parties. These instruments were developed to provide a national legal infrastructure to prevent nuclear accidents and to mitigate their consequences should they happen.

An equally important enhancement of nuclear security is provided by instruments related to safeguards, such as the NPT, Safeguards Agreements, Additional Protocols and nuclear supplier controls. These instruments were developed to restrain State activities towards weapons development.

The revised CPPNM emphasizes States' responsibility for the physical protection of nuclear material and facilities on their territory; for domestic and international transport of nuclear material; for protection against sabotage; for protecting confidential information; and sets the physical protection objectives and fundamental principles and punishable acts that States must prosecute. It also defines the categories of nuclear material and corresponding levels of protection of nuclear material in international transport.

In essence, the physical protection objectives [34] are to protect against theft, locate and recover stolen material, protect against sabotage and mitigate the radiological consequences of sabotage. The fundamental principles of a State Physical Protection System relate to State responsibilities during transport, for establishing a legislative and regulatory framework, for establishing a competent authority, for specifying the responsibilities of the licence holder, for creating a security culture, for adopting a threat based approach, a graded approach, defence in depth, quality assurance, contingency plans and confidentiality.

UN Security Council Resolution 1540 is binding on Member States; it focuses on preventing the proliferation of weapons of mass destruction, including nuclear weapons. It specifically addresses concerns over terrorism and illicit trafficking, obliges all States to take and enforce effective measures to prevent proliferation and specifically references the need to develop and maintain appropriate physical protection measures and to account for nuclear material.

The Nuclear Terrorism Convention was opened for signature in September 2005 and is the most recent of 13 UN anti-terrorism instruments. It has a definition of acts of nuclear terrorism and a broad definition of radioactive material, so that it covers RDDs as well as nuclear explosive devices. It criminalizes a wide range of activities involving nuclear or radioactive material. It overlaps with other instruments, particularly UNSCR 1540 and the CPPNM. It also includes obligations to cooperate, share information and inform the UN Secretary General and the IAEA, and an obligation to protect radioactive materials, taking into account IAEA recommendations and functions.

6. IAEA RESPONSE TO SECURITY THREATS

The IAEA serves as a central component of the international security infrastructure, which provides the framework for cooperation. The IAEA has prepared, in cooperation with Member States, a Nuclear Security Plan 2006–2009 [35], that was approved by the IAEA Board of Governors and endorsed by the General Conference in September 2005 to address threats to nuclear security. It is the continuation and expansion of the initial plan adopted in 2002 [36]. The IAEA's Office of Nuclear Security is in charge of coordinating and implementing the plan. The plan is divided into three areas: coordination and data analysis (including the Illicit Trafficking Database), prevention, and detection and response. It also includes activities from the areas of safety and safeguards that contribute to security. The main activities include promoting the international instruments relating to nuclear security, establishing international nuclear security recommendations and guidance publications, performing activities related to human resource development (training activities and workshops), and providing nuclear security services, in which the needs of States are identified and addressed in Integrated Nuclear Security Support Plans. It also covers the security upgrading of nuclear facilities, facilities with radioactive sources and in relation to the combating of illicit trafficking, such as border monitoring equipment.

7. GUIDELINES AND RECOMMENDATIONS

One of high priorities of the IAEA is establishing nuclear security guidance for States on the implementation of the conventions and providing supplementary measures for nuclear security at the State, regulator and operator levels. The best known document with recommendations on physical protection is INFCIRC/225/Rev. 4 (The Physical Protection of Nuclear Material and Nuclear Facilities), which originated in 1972, when it was first published as the 'Grey book'. INFCIRC/225 establishes three categories of nuclear material (see Table 1) as a function of proliferation risk and recommends graded measures of physical protection against theft for each category of nuclear material in use, in storage or in transport. It also covers protection against sabotage. In Table 1, for comparison, values are shown of significant quantities of nuclear material (SQ), that is, the amount of material

TABLE 1. CATEGORIES OF NUCLEAR MATERIAL AND SQs

Nuclear material	1 SQ	Category		
		1	2	3
Pu or U-233 unirradiated	8 kg	>2 kg	0.5–2 kg	0.015–0.5 kg
HEU >20%	25 kg	>5 kg	1–5 kg	0.015–1 kg
LEU >10%			>10 kg	1–10 kg
LEU <10%				>10 kg
Irradiated fuel			DU, natural U, Thorium, LEU	

for which the construction of a simple nuclear explosive device cannot be excluded. These values are not included in INFCIRC/225/Rev. 4.

For a typical nuclear power plant site, the level of security measures is not driven by the fresh fuel (which is only category 3 material), but rather by the fact that there is irradiated fuel on-site (category 2 material) and by the need for protection against sabotage. Some of the measures for category 2 material include establishing protected areas inside the controlled area, having a hardened central alarm station inside the protected area that is continuously manned with redundant and diverse communication with management, response forces off-site and guards on-site, the continuous presence of guards, uninterrupted power supply for intrusion sensors, lighting and communications. Each area should be provided with a barrier. Intrusion detection should be performed at the physical barrier surrounding the protected area, and timely assessment should be carried out. All intrusion detection sensors should annunciate and be recorded in the central alarm station. Clear areas should be provided on both sides of the perimeter of the protected area with sufficient illumination for assessment. Access to protected areas should be limited to trustworthy or escorted persons only upon identification and issuance of a badge. All vehicles, persons and packages entering or leaving protected areas should be searched. Keys and card key records and control should be established. Employee security awareness training should be done on a yearly basis. Contingency plans should be established and tested. For protection against sabotage, an additional recommendation is to establish vital areas to enclose or shield all systems, structures and components that prevent the release of substantial amounts of radioactivity to the environment. Vital areas should be located inside protected areas. INFCIRC/225 and related guidance publications [37] provide recommendations and further practical advice [38] for Member States.

Internationally accepted baseline publications dealing with nuclear security are now being developed in the new IAEA Nuclear Security Series, covering nuclear and other radioactive materials, and associated facilities and transport. The process for the development of the Nuclear Security Series publications was established to ensure high quality, consistency with other IAEA standards and guidance, and broad international consensus through the involvement of Member States. The Fundamentals of Nuclear Security will represent the top level, and Recommendations the second level. The third level will comprise different implementing Guides. These will provide guidance on areas such as design basis threat, protection against sabotage, vital area identification (e.g. the spent fuel pool might be considered as a vital area), protection against insider threats, identification of radioactive sources, security of radioactive sources, security of radioactive waste and transport security, and on combating illicit trafficking. Also, several cross-cutting areas will be covered, such as nuclear security culture, information technology security, confidentiality of information and emergency response guidance. The first three publications in the Nuclear Security Series, covering specifications for border monitoring equipment, detection of radioactive material in public mail and nuclear forensics, were recently published and are available on the IAEA's web site.

8. EVALUATION AND ADVISORY SERVICES

Different nuclear security missions, evaluations and technical visits are the IAEA's main tool for assisting States in improving nuclear security by identifying nuclear security needs. These can subsequently be addressed by the State alone, or addressed in conjunction with IAEA support, funded through the voluntary Nuclear Security Fund, or addressed with the assistance of a bilateral partner.

Most relevant to spent nuclear fuel security are the International Physical Protection Advisory Service (IPPAS) missions. They continue to serve as the IAEA's main tool for evaluating existing physical protection arrangements in Member States. In IPPAS missions, detailed reviews are carried out of the legal and regulatory basis for the physical protection of nuclear activities in the requesting State and of compliance with the obligations contained in the CPPNM and with the guidance provided in INFCIRC/225/Rev. 4, as well as with international best practices. The findings of IPPAS missions are presented in confidential mission reports for further action. Specific IPPAS follow-up assistance such as training, technical support and more targeted assessments continue to constitute an essential feature of this advisory service.

The International Nuclear Security Service (INSServ) mission serves as a flexible mechanism to help identify a State's broad nuclear security requirements and the measures needed to meet them.

International Team of Experts (ITE) advisory missions are a primary mechanism to advise States regarding their adherence to or implementation of international instruments relevant to enhancing protection against nuclear terrorism.

9. EDUCATION AND TRAINING

To assist States in establishing and maintaining effective physical protection of nuclear and other radioactive material, a variety of training courses and workshops at the international, regional and national levels is offered. The target audience depends on the subject of the course or workshop. It can be policy makers, nuclear regulators, facility operators, legislators, emergency responders, police, customs, border forces, or military and intelligence services.

The topics include basic security objectives and fundamental principles, basic and advanced understanding of physical protection, and a systematic methodology to design and evaluate physical protection systems for nuclear facilities that are effective against theft and sabotage. Practical exercises and nuclear facility tours are important elements of such training events.

To strengthen security arrangements for radioactive sources, further attention is focused on physical protection and control of radioactive sources throughout their life cycles. Specialized physical protection courses include national workshops on the methodology to develop the design basis threat required to define the performance targets for physical protection systems, a course on the technical features of physical protection systems which includes hands-on training, and a course to prepare national authorities for conducting inspections of physical protection arrangements.

10. TECHNICAL IMPROVEMENTS AND UPGRADES

The IAEA is assisting States in upgrading physical protection systems for materials and facilities that were identified through INSServ missions. This can be done by bilateral support from Member States or to some extent by the IAEA, using the Nuclear Security Fund.

11. CONCLUSIONS

The threat of nuclear terrorism is real. Potential targets are abundant and spent fuel cannot be excluded. Nuclear security is the responsibility of States, but is also subject to the emerging international nuclear security regime. States are addressing the threats to nuclear and radioactive material, associated facilities and transport. In the area of spent fuel, several studies have been carried out or are ongoing to assess consequences and vulnerabilities. Different measures have been taken to reduce the threats and vulnerabilities, including design basis threat reassessment and physical protection enhancements. In parallel, other measures have been taken to protect major infrastructures in States against terrorist attack, including aviation security enhancement and information sharing, while taking care to protect confidential information at the State and international levels. All this contributes to nuclear security. To further improve nuclear security worldwide, it is necessary to strive for universal adherence to international nuclear security related legal instruments and their implementation, including continuing use of the IAEA's nuclear security advisory services, to produce the IAEA's nuclear security guidance publications and to work on their implementation — based on Member States' needs. Finally, it has to be recognized that safety, safeguards and security are all prerequisites for nuclear power sustainability and renaissance.

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DISCUSSION

R. EINZIGER (USA): In the nuclear security area, how can one define a design basis threat when one has no idea of what is going on in the minds of potential terrorists?

M. GREGORIC (IAEA): Each State is responsible for its nuclear security and it is up to the State to arrive at the final design basis threat. The IAEA is helping by producing a guidance publication on the methodology to use in defining the design basis threat. To answer your question more directly, there are certain threats that can be protected against by the plant itself and there are certain threats that, of course, no matter what protection the plant provides, cannot be protected against. So there is an area that States must try to resolve themselves, on many different levels.

INTERNATIONAL SAFETY REGIME

(Session 1.B)

Chairperson

A. GONZÁLEZ
Argentina

IMPLEMENTATION OF THE JOINT CONVENTION ON THE SAFETY OF SPENT FUEL MANAGEMENT AND ON THE SAFETY OF RADIOACTIVE WASTE MANAGEMENT

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Abstract

The paper gives a brief history of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (the Joint Convention) and explains its purpose, its main content and its method of operation. The progress achieved to date is summarized, with particular reference to the results of the second review meeting held in May 2006. Finally, consideration is given to the future prospects of the Joint Convention and, in particular, of the need to attract a greater number of Contracting Parties and to improve the mechanisms of the Convention to ensure that it remains an effective instrument for the improvement of the safety of spent fuel and radioactive waste management in the world.

1. BACKGROUND: HISTORY OF THE JOINT CONVENTION

The Convention on Nuclear Safety (CNS) and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (the Joint Convention) were agreed to in the aftermath of the Chernobyl accident of 1986. They were drafted based on some general ideas:

(a) Nuclear safety is a global issue.
(b) Nuclear safety should be improved and should gain visibility.
(c) After the experience of the Chernobyl accident it was recognized that national safety authorities in some countries were not strong enough and that there was a need to better define the responsibilities and functions of the regulatory bodies and of the allocation of duties, to improve the interaction between regulators and licensees and to improve the ways in which regulators generally communicate and provide information about the performance of their duties.

(d) It was necessary to find a way to improve communication between countries and, in particular, between nuclear safety authorities. A number of senior regulators and operators felt the need for a forum where they could exchange views and opinions. With the background of the Chernobyl accident, there was a particular need for a forum at which national safety authorities from Eastern European countries could exchange views on the effectiveness and efficiency of their regulatory systems.

The first Convention to be created after the Chernobyl accident was the Convention on Nuclear Safety, which was adopted in 1994 and entered into force in 1996. During the discussions on the establishment of this convention it was generally recognized that nuclear safety is a common issue for many countries but that the safety of the management of radioactive waste is also a common issue. The safety of the management of spent fuel was also identified as an issue. Thus there was a clear recognition that after establishment of the Nuclear Safety Convention, there would be a need for another convention on the safety of radioactive waste management and/or on the safety of the management of spent fuel. One of the difficult issues was the status of spent fuel. For some countries spent fuel is a waste not to be used for any other purpose, but for others spent fuel is a raw material with the potential for re-use through, e.g. reprocessing. Many meetings were necessary for the issue to be fully addressed and for a consensus to be found. This was finally achieved by avoiding the issue in the convention. Instead, a single convention addressing two parallel subjects was created, radioactive waste and spent fuel. What is called the 'Joint Convention' therefore includes a number of similar articles, but in one part of the convention they are devoted to spent fuel and in the other part to radioactive waste. The Joint Convention was adopted in 1997 and entered into force in 2001.

2. OBJECTIVES, NATURE AND MAIN MECHANISMS OF THE JOINT CONVENTION

The CNS and the Joint Convention are 'incentive' in nature. Their intent is primarily to provide a tool to help the Contracting Parties make continuous progress. More specifically, the objectives of the Joint Convention are:

(a) To achieve and maintain a high level of safety worldwide in spent fuel and radioactive waste management through the enhancement of national measures and international cooperation;

(b) To ensure effective defences against potential hazards so that individuals, society and the environment are protected from the harmful effects of ionizing radiation now and in the future;
(c) To prevent accidents with radiological consequences and to mitigate their consequences should they occur.

To achieve this, the main general provisions of the Joint Convention are that:

(1) Each Contracting Party shall take all necessary legislative, regulatory and administrative measures for implementing its obligations;
(2) Each Contracting Party shall establish and maintain a legislative and regulatory framework, taking into account the objectives of the Convention;
(3) Each Contracting Party shall establish a regulatory body with adequate authority, competence, financial and human resources and with effective independence;
(4) Each Contracting Party shall ensure that the prime responsibility for spent fuel and radioactive waste management rests with the licence holder.

Keeping in mind the initial idea of creating a forum for the exchange of experience and the exchange of views on problems and difficulties, the main operational part of the Convention is, in practice, a peer review mechanism. This means that the mechanism to facilitate exchanges between Contracting Parties on the implementation of the Convention is participation at the review meetings (to be held regularly). Each Contracting Party has to submit a report presenting the measures it has taken to implement the obligations of the Convention. Each national report is sent in advance to all other Contracting Parties, who may ask questions to obtain further details or clarifications. At the review meeting, each report is presented and discussed in a country group meeting (at the second review meeting there were between 6 and 9 countries per country group). At the end of the process, during a plenary session involving all Contracting Parties, a summary report addressing the main issues discussed during the country group sessions and presenting the conclusions of the review meeting is adopted by consensus and made publicly available.

In summary, the Joint Convention is an incentive convention. When a country first becomes a Contracting Party it accepts a number of new obligations, but the main one is to make a report to the review meeting every 3 years, to present it, to defend it, and to participate in and learn from the general discussion.

3. PROGRESS ACHIEVED AT THE SECOND REVIEW MEETING

The number of Contracting Parties has increased since the first review meeting. At the first review meeting in November 2003 there were 33 Contracting Parties and at the second review meeting in May 2006 this had increased to 41 Contracting Parties. The Russian Federation was a full participant at the second review meeting and China was allowed to attend the meeting since it is finalizing its accession as a full Contracting Party. Now, almost all countries with nuclear power plants are Contracting Parties, with the exception of Armenia, India, Mexico, Pakistan and South Africa. It should be recognized that almost every country in the world produces radioactive waste. Thus, although countries with no nuclear power plants but with nuclear applications such as the use of isotopes in medicine should also be concerned, only a limited number of them have so far become Contracting Parties.

All the participants presented comprehensive views of their national situations. The discussions were intense and fruitful, even on difficult issues. In terms of topical issues addressed, the main progress achieved is as follows. A number of countries are developing strategies for spent fuel and radioactive waste management. A number of countries are establishing long term policies and are trying to find better ways of managing spent fuel and radioactive waste. Compared to the situation at the first review meeting, national strategies have been further developed and progress has been made in their implementation, although there is a wide range in the stages of implementation. Increasing importance is being given to public consultation and the need for public acceptance is being increasingly recognized. Progress was also identified in the development of funding strategies. Another topic discussed, which is relevant for all countries, is the issue of disused radioactive sources. There were, in particular, clear indications that long term management policies are being established. There were fruitful exchanges on registration processes, funding schemes and on the implementation of the IAEA Code of Conduct on the Safety and Security of Radioactive Sources.

One of the major difficulties remains the implementation of geological disposal of radioactive waste. A number of countries envisage this solution. Some progress has been made in several countries but generally it remains a difficult matter, mainly because of the very long term issues involved. In this regard, and for the first time, there was a discussion on the idea of regional repositories. But the difficulties remain, since those countries which expressed interest in regional repositories also expressed the wish that they should be located outside their national territories.

4. PROGRESS EXPECTED AT THE THIRD REVIEW MEETING

The first issue is still to increase the number of participating countries in the Joint Convention. As indicated above, some countries with nuclear power plants and many other countries without nuclear power plants but with nuclear applications and facing radioactive waste management challenges are not yet Contracting Parties. In this regard, it is important to realize that the main obligation in becoming a Contracting Party is participation in the review process.

The second issue is to improve the review process and to make it more effective. Clearly, the peer review process should not be changed for almost all Contracting Parties that participated in the second review meeting. That means that the concept of national reports addressing all issues related to spent fuel and radioactive waste management should be retained. The proposal for establishing topical sessions as part of the review meeting was discussed, but there was no consensus on it. One of the ideas discussed for the improvement of the review process was to avoid, in future national reports, too much focus on legal and regulatory issues. It was suggested that while retaining the concept of self standing national reports, they should focus more on details concerned with the practical implementation of the actions decided upon and the experience feedback gained.

Another issue discussed relates to the frequency of the review meetings. For countries that are parties to both the Joint Convention and the Nuclear Safety Convention the drafting of national reports, the preparation for and the participation at the review meetings is a significant burden. Countries participating in both conventions have staff members continuously and deeply involved in the preparation of a national report either for the Joint Convention or for the Nuclear Safety Convention. One of the proposals discussed was to adopt a frequency of four years for the review meetings. A comparison was made with the olympic games, with alternate summer and winter games, but this is still an open issue.

Finally, there was a discussion of the role of the IAEA Safety Standards in the review process, and a wide range of views was expressed. Some countries considered that the IAEA Safety Standards should be used as the basis for the preparation of national reports in order to demonstrate the implementation of the Joint Convention. Others considered that it is up to each country to take a decision on how the IAEA Safety Standards should be used, and that they should be seen as ‘a useful source of guidance’ in this context.

DISCUSSION

A. GONZÁLEZ (Argentina — Chairperson): I should like to ask a question that I am often asked by non-experts. As you said, the Joint Convention is an ‘incentive’ convention, but these people say “An incentive convention is fine, but with other conventions the parties really have to comply with specific provisions. For example, if an airline does not comply with the ICAO Convention it may be banned from using certain airports.

They say that the parties to an incentive convention are simply members of a club who meet every few years, tell one another what they are doing and then go home happy that the situation is OK. At the latest Review Meeting of Contracting Parties to the Joint Convention the representatives of one country even indicated that in their country there was radioactive waste which was simply being put into the ground, and nobody criticized that practice. So some people want to know what assurance the Joint Convention gives that the parties really are going to comply with some safety limits or constraints which ensure that they are doing things properly.

A.-C. LACOSTE (France): Essentially you are asking about what the effectiveness of an incentive convention can be. My first answer is this — if the Joint Convention were not an incentive convention but had some mandatory aspects a number of countries that are Contracting Parties would not have acceded to it. I do not think that would have been a good thing. My second answer is this — not only because of the peer review process, but because we know one another, countries have quite a good idea of what is happening in other countries. For example, the French nuclear safety authority has a good idea of what is happening in 30 or 40 other countries as far as radioactive waste and spent fuel are concerned. In my opinion, a country cannot tell lies in a process like the Joint Convention peer review process.

A further question concerns the openness of the review process. The question has often been asked, “Why don’t you allow civil society, journalists, people from environmental organizations to participate?” My answer is, “Of course, we could open the review meetings to them, but then quite a lot of countries would not send representatives to the review meetings and quite a lot would decide not to become Contracting Parties in the first place”, or put another way, “If we want open mouths, we must have closed doors”. But the two questions — the opening up of participation and the incentive aspect of the Joint Convention — are closely linked. Up to now, a choice on these questions has been made, but at each review meeting we should ask ourselves whether there has been real progress.

A. GONZÁLEZ (Argentina — Chairperson): In his last slide, Mr. Lacoste posed a key question concerning the role of the international safety standards in the Joint Convention. Let me remind you of one important point: in the Convention on Nuclear Safety there is no reference to any standards, but in the preamble to the Joint Convention there is a reference to the international safety standards. So in this respect there is a substantive difference between the Convention on Nuclear Safety and the Joint Convention.

A.-C. LACOSTE (France): Allow me to add something to my first presentation. I participated in the first two Review Meetings of Contracting Parties to the Convention on Nuclear Safety. The first review meeting was a difficult meeting, while the second one was perceived as being successful. The third review meeting — so it seemed to me at least — was perceived as being disappointing. We are not sure what will happen at the third Review Meeting of Contracting Parties to the Joint Convention, the danger being that people are becoming too clever at presenting things in a favourable light. We shall see.

TOWARDS A CONSISTENT SET OF HARMONIZED SAFETY STANDARDS

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Abstract

The development of the IAEA Safety Standards is overseen by a committee of senior regulators, the Commission on Safety Standards (CSS). This paper describes the recent work of the CSS in restructuring the IAEA Safety Standards by creating a single publication to head the series, a Safety Fundamentals, in place of the three publications, each covering a different safety theme, which currently head the series. The new publication will provide a common philosophical safety basis for the future development of publications in all four thematic areas: nuclear, radiation, waste and transport.

1. BACKGROUND: THREE SAFETY FUNDAMENTALS HEADING THE SAFETY STANDARDS SERIES

According to Article III.A.6 of its Statute, the International Atomic Energy Agency (IAEA) is “authorized to establish or adopt…standards of safety for the protection of health and minimization of danger to life”. Ten years ago, four Safety Standards programmes existed in parallel in the IAEA, on Nuclear Safety, Radiation Safety, Waste Safety and Transport Safety. Also ten years ago the IAEA’s Board of Governors (BoG) approved the publication of a third Safety Fundamentals publication. Prior to this, the BoG had approved Safety Fundamentals on the Safety of Nuclear Installations in June 1993 and on the Principles of Radioactive Waste Management in March 1995. The third one, approved in June 1995, was on Radiation Protection and the Safety of Radiation Sources. It was also meant to cover indirectly the safety of the transport of radioactive materials.

These publications are quite different in nature. The Safety Fundamentals on the Safety of Nuclear Installations includes one general nuclear safety objective, one radiation safety objective, one technical safety objective and 25 safety principles. It served as a basis for the technical obligations of the Convention on Nuclear Safety. The Safety Fundamentals on the Safety of

Radioactive Waste Management includes one radioactive waste management objective and 9 fundamental principles. It served as a basis for the technical obligations of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The third Safety Fundamentals publication, on Radiation Safety, included 1 protection safety objective, 1 safety objective and 11 fundamental principles. It served as a basis for the Code of Conduct on the Safety and Security of Radiation Sources. This meant that there were three different and not necessarily coherent Safety Fundamentals as the lead publications heading the whole collection of IAEA Safety Standards.

Thus, when approving this third Safety Fundamentals publication in June 1995, the BoG realized that the philosophies for nuclear safety, radiation safety and waste safety were technically compatible but expressed differently and therefore requested the IAEA Secretariat to consider, at an appropriate time, the revision of the three Safety Fundamentals publications with the aim of combining them, with the objective of establishing a common and coherent safety philosophy. This led to a report prepared by the International Nuclear Safety Advisory Group (INSAG) in 1999 on how to establish a coherent and unified approach to safety philosophy. In its preamble, the IAEA Director General, Mohamed ElBaradei, stated that "This report intends to show that, at the conceptual level, the distinction traditionally made between nuclear safety and radiation protection is hardly justifiable".

2. WORK TOWARDS HARMONIZATION: INVOLVEMENT OF THE COMMISSION ON SAFETY STANDARDS AND THE COMMITTEES

The work towards harmonization started with the creation of the Department of Nuclear Safety of the IAEA and the establishment of the Commission on Safety Standards (CSS), together with four specialized Safety Standards Committees (dealing with radiation safety, nuclear safety, transport safety and waste safety) for the review and approval of publications. With the help of the Commission and the Committees, a harmonized process was put in place for the preparation and review of the IAEA Safety Standards and the interrelationship between the different subject areas was addressed, mainly through the involvement of several Committees for some of the publications and, in many cases, joint meetings of the Committees.

The work of the CSS and the four committees dealing with radiation safety, nuclear safety, transport and waste safety, resulted in significant progress. In 1999 the CSS approved a Document Preparation Profile for the development of a single Safety Fundamentals, in which it:

(a) Stressed that the publication should address protection and safety as a single concept;
(b) Recommended that the publication should be as universal as possible;
(c) Insisted that all Safety Standards committees be involved in the development and review.

A first outline was prepared at the end of 2000 and submitted to the Committees. Reports made to the CSS at the end of 2000 indicated that difficulties were being encountered due to the risk of loss or dilution of the contents of the current set of Safety Fundamentals. Therefore, in 2001 the CSS proposed to the Director General of the IAEA that a strategy should be prepared for the IAEA Safety Standards programme aimed at enhancing the IAEA Safety Standards and achieving their global recognition. At the end of 2002 the CSS set up a small working group which created a new draft of the common Safety Fundamentals. In the period 2002–2003, as part of the strategy and action plan for the development and application of the IAEA Safety Standards, a new approach based on simplification was adopted. Recognizing that the existing Safety Fundamentals included many requirements now covered in the next tier of publications, the Safety Requirements, the idea was to restrict the contents of the Safety Fundamentals to common and basic principles and to address the key fundamental concepts through these.

This resulted in a new draft being submitted to IAEA Member States for comment at the end of 2004, reviewed by the four Safety Standards Committees in 2005, and finally approved by the CSS in June 2006 as a single Safety Fundamentals publication, with 10 principles common to all areas of safety. The publication is written in such a way that it can be understood by the stakeholders and the public. More particularly, the objective was that any interested member of Parliament or Government should be able to understand the basic safety principles. This is why the publication is written without technical jargon and in plain words.

3. NEXT STEPS: PUBLICATION OF THE SINGLE SAFETY FUNDAMENTALS AND ESTABLISHMENT OF A HARMONIZED SET OF SAFETY REQUIREMENTS AND GUIDES

The next steps are the submission of the draft Safety Fundamentals publication to the IAEA Board of Governors for approval as an IAEA publication and its submission to the governing bodies of the several other co-sponsoring international organizations for approval. When the new Safety Fundamentals have been approved there will finally be a basis for a consistent set of safety publications to be used by Member States. This single publication and the subsequent set of Safety Requirements and Guides will:

(a) Implement the concept of 'safety as a whole', as defined by the Convention on Nuclear Safety;
(b) Provide a common language for a common understanding of radiation and nuclear safety;
(c) Help the implementation of the Strategy for the IAEA Safety Standards towards a global reference point for nuclear safety, radioactive waste management, radiation protection and for the transport of radioactive material.

There are, however, still a number of different views on this latter objective: should the IAEA Safety Standards be used as a 'reference'? Should they be used as a 'tool'? Should they be used as a 'useful tool'? Should they be considered as a 'source of guidance'? There was a considerable amount of discussion at the end of the second review meeting of the Joint Convention on this and on the use of the IAEA Safety Standards by Member States in general and on their use in the context of the review meetings of the international conventions in particular. Some examples were given of the direct use of the IAEA Safety Standards as national regulations. Several other examples were given of their use as a reference for drafting national regulations. This latter case is quite similar to the approach currently being followed by WENRA (Western European Nuclear Regulatory Association), which uses the IAEA Safety Standards for harmonization purposes.

DISCUSSION

A. GONZÁLEZ (Argentina — Chairperson): We have a convention and we will have a set of international standards, which may help us to harmonize safety in this complicated area. Let me underline what Mr. Lacoste said — that this approval of a single set of Safety Fundamentals will really be a revolutionary action in the history of the international IAEA Safety Standards. The present separation into three sets of Safety Fundamentals is not helpful; it creates unevenness and even some contradiction. It took a lot of effort, particularly on the part of Mr. Lacoste, to bring about this change.

T. TANIGUCHI (IAEA): Yesterday there was a lot of interesting discussion about recent initiatives — national (those of the USA and the Russian Federation) and international. However, not much was said about international legal instruments and safety standards. The policy discussions about establishing, for instance, multinational approaches tend to focus more on the promotional issues, for obvious reasons, but I have been wondering whether Mr. Lacoste has any suggestions on how to link the safety area, particularly the international legal instruments and safety standards, with these current discussions and initiatives.

A.-C. LACOSTE (France): This is a difficult issue — what is the official or legal status of the IAEA Safety Standards? Sometimes different representatives of the same country do not take the same position; that has been so in the case of the USA. Depending on the topics being discussed, depending on the convention, depending on the way the delegation is led, the position of the country is not the same.

The issue is clear: do we wish to give a formal regulatory status to the IAEA's Safety Standards? The answer is not obvious because, should we decide to give that kind of status to them, the work on establishing them would not be the same and I do not know what quality of standards would be attained. My personal feeling is that this is such a difficult topic that we have no other way forward than first to continue our efforts to produce safety standards of as high a quality as possible. In that connection we should bear in mind that historically many of the IAEA Safety Standards represented, at one time, the lowest common denominator. Then they became collections of good practices and now they are collections of good practices and sometimes of best practices. So we should continue our efforts to improve the quality of standards. Second, we should continue expanding the scope of the standards. For spent fuel there is still a need for additional standards. Third, we should start trying to persuade people and countries that the use of safety standards is a good thing for them. But this is obviously a long term process.

What I cannot imagine is any kind of global decision that we now have a new safety regime involving the mandatory use of standards. So, I think what is required is long term persuasion with ongoing efforts to improve quality.

A. GONZÁLEZ (Argentina — Chairperson): I believe that the point of Mr. Taniguchi was somewhat independent of this legal status issue, which — as you indicate — is controversial and difficult. The point was that in the area of spent fuel management the common theme has been that of international partnerships, which does not arise in other areas. In this area, the expression 'international partnerships' is used repeatedly but it is difficult to understand how you can have an international partnership covering everything except safety. It seems to me that the idea of international partnerships should be linked with the idea of international harmonization of the IAEA Safety Standards to be applied in that partnership. That, I believe, is Mr. Taniguchi's point, and it is somewhat independent of the legal issue.

A.-C. LACOSTE (France): Regarding the establishment of a single set of Safety Fundamentals, I believe that it is a way of bringing nuclear safety and radiation protection closer together. For me, this is the beginning of a real merger of radiation protection and nuclear safety. I am the chairman of the International Nuclear Regulators Association, which consists of the heads of the nuclear safety authorities of eight countries (Canada, France, Germany, Japan, Spain, Sweden, the UK and the USA). Up to now we have been dealing only with nuclear safety issues. At our last meeting (held in Paris in March), for the first time we started discussing radiation protection issues, and our feeling was that there should be no separation between the two areas. At our next meeting (to be held in Avignon in September), L. Holm, Chairman of the International Commission on Radiological Protection (ICRP), will give a presentation about the ICRP's work. I think this is really the beginning of a new era, without a separation between the four domains — radioactive waste safety, radiation safety, nuclear safety and transport safety.

J. BOUCHARD (France): Regarding the initiatives we discussed yesterday, we are trying to devise multilateral processes for the nuclear fuel cycle, new types of reactor and so on, and we shall have to adapt the IAEA Safety Standards to them. I have been involved in discussions of this within the framework of the Generation IV International Forum. Within that framework we have tried, with only limited success, to discuss with safety authorities how to develop standards that take account of the future new technologies. The aspect of applicable safety standards was not completely clear in the presentations on the new initiatives.

A. GONZÁLEZ (Argentina — Chairperson): Mr. Lacoste made a very important point — the new Safety Fundamentals are really fundamental. They are different from what we had before — what we used to call 'Fundamentals' were not really fundamental; for example, some of them applied to a particular type of reactor. So this may be the beginning of a new era, as Mr. Lacoste said, and the Safety Fundamentals can serve as a basic framework for these recent initiatives, and for international partnerships.

TRANSPORT ISSUES

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Abstract

The trends in fuel management include higher discharge burnup and long term storage at the reactor site prior to final disposal. These strategies present technical issues which should be addressed before the fuel is discharged and stored so that appropriate storage regimes can be put in place to minimize issues that may arise at the time of their future transport. All stakeholders have a role to play in providing a viable and effective transport route in the future. This requires that both fuel designers and reactor site operators have an understanding of the requirements of the transport safety cases and the regulatory regime in which they exist. The continuous development of technical solutions and of regulatory compliance frameworks and the sharing of best practices throughout the industry are vital to ensure that the transport of long stored spent fuel is carried out within known and intended safety margins.

1. INTRODUCTION

In line with the scope of the conference, this paper focuses on the technical issues surrounding the transport of spent fuel. Political, security and safeguards issues are not included. The issues raised are based upon the premise that higher burnup and longer irradiation cycles are being sought by reactor operators to reduce down times over the lifetime of a reactor to a minimum. It is recognized that there are many factors which influence the operating profile of a reactor, but it is likely that the 'transportability' of the spent fuel being discharged from a rector core is low on the list of priorities.

2. GLOBAL PERSPECTIVES — NUCLEAR POWER GENERATION AND TRANSPORT EXPERIENCE

According to data published by the International Atomic Energy Agency (IAEA) [1] there are 440 nuclear power reactors (NPRs) operating in the world in 31 countries, with an additional 26 NPRs currently under construction. To provide an appreciation of the amount of spent fuel involved, the operating

NPRs contain approximately 30 000 tonnes of uranium fuel within their reactor cores. Assuming a reload of one third of the reactor core fuel every 18 months, this represents 7000 tonnes of spent fuel arising each year.

To respond to this current and future transport demand the nuclear industry looks to the past experience of spent fuel transport. In some countries, spent fuel has been routinely transported for over 35 years by road, rail and sea, with some routes involving more than one mode of transport. Overall, this represents many millions of kilometres travelled and several thousand spent fuel cask journeys. While such statistics are often cited by industry (and challenged by some groups) they do provide evidence of the maturity and effectiveness of the infrastructures operated throughout the world. To date, there has never been a release of radioactive material from a spent fuel cask during transport. A typical spent fuel transport cask can weigh between 30 and 120 tonnes, and each can cost up to €1.4 million.

3. BASIS OF SUCCESS

There are several elements that have contributed to the safety record of spent fuel transport over the past four decades, including:

(a) The evolution and continual review of the IAEA Transport Regulations and guidance publications, which have:
   — Acted as a focal point of expertise in the spent fuel transport sector,
   — Engendered a robust safety culture within the industry,
   — Provided opportunities for all stakeholders to be involved in the review process and to become aware of scientific and technical advances and the experience of those involved in the transport of radioactive material;

(b) An effective nuclear transport community that includes competent authorities, government departments, industry, academia and non-governmental organizations;

(c) The adoption of the IAEA transport regulations by IAEA Member States and their transposition in national legislative frameworks by various mechanisms, enabling them to be enforceable by law.

4. REGULATORY STRUCTURE

The IAEA Safety Standards Series No. TS-R-1 [2] forms the basis of transport legislation for radioactive material in the 137 Member States of the

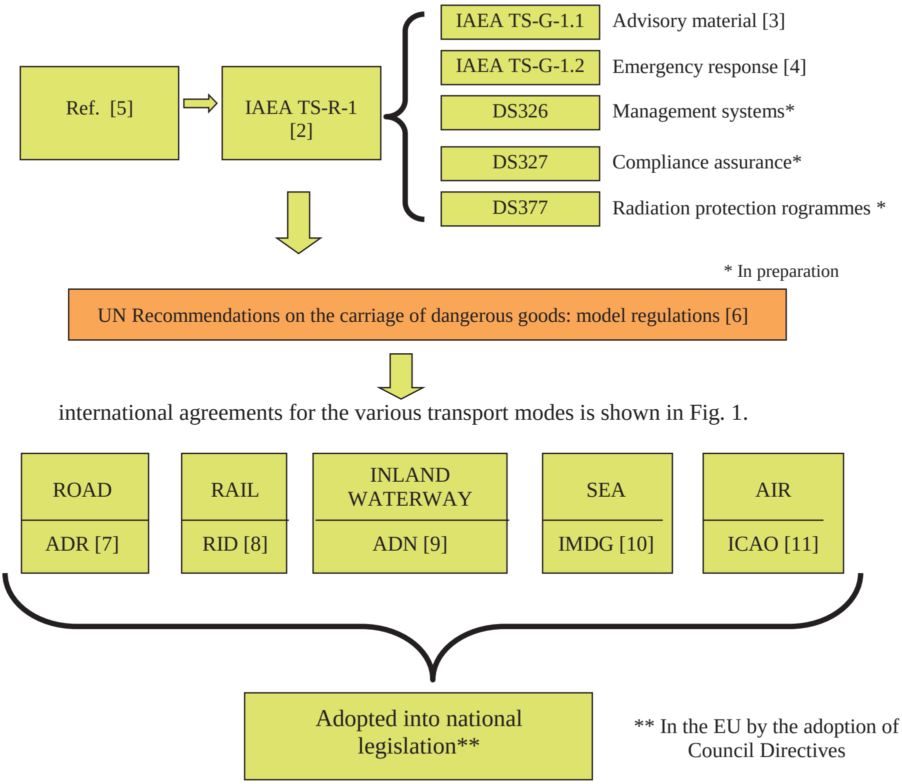

FIG. 1. Relationship between IAEA regulations, the UN Model Regulations and the international agreements for the various transport modes.

IAEA. However, as stated in §103 of Ref. [2], the actions and requirements are prescribed but the responsibilities are not specifically assigned to any legal person. The objective of the regulations is to protect persons, property and the environment from the effects of radiation during the transport of radioactive material ([2] (§104)).

The relationship between the IAEA regulations, the UN Model Regulations and the international agreements for the various transport modes is shown in Fig. 1.

5. INTERNATIONAL SAFETY REGIME

Four key elements of the international safety regime are: establishing requirements; adopting them into national law; providing powers of enforcement to competent authorities; and encouraging interaction between stakeholders to develop and share best practices. Each element is further discussed below.

5.1. Establishing requirements

The IAEA Safety Requirements and Safety Guides provide a basis upon which legislation at a national level can be formulated. There should be a clear recognition throughout the documentation of the relationship between storage (ponds, dry store and cask), transport and the requirements of receiving facilities, so as to provide mechanisms which ensure that the requirements for transport are considered throughout the spent fuel management process. Emphasis has to be maintained on providing explanatory and guidance documentation to ensure that variations in interpretation of the regulatory requirements are minimized.

5.2. Adopting requirements in national law

The IAEA transport requirements with assigned responsibilities (for example consignor, carrier) are adopted in national legislation. In the context of spent fuel this can be considered within three sectors, namely nuclear power plant (storage), transport in the public domain, and the receiving facility (reprocessing plant, decay store or disposal facility).

The scope of transport covers all operations and conditions associated with and involved in the movement of radioactive material, including the design, manufacture, maintenance and repair of packaging (in this instance a spent fuel cask), and the preparation, consignment, loading, carriage (including in-transit storage), unloading and receipt at the final destination of loads of radioactive material and packages ([2] (§106)).

The three sectors of the spent fuel transport model can often be the responsibility of more than one government department and it is therefore vital that the requirements for transport are acknowledged accordingly. Part of this acknowledgement would be the clear definition and assignment of responsibilities.

5.3. Providing powers of enforcement to competent authorities

Appropriate powers of enforcement should be provided in law to enable the regulatory requirements to be enforced by the competent authority or its agent. It would be prudent if the powers provided a basis for various levels of intervention by a competent authority, ranging from requiring an organization to improve its package designs/quality programmes/systems of working, so as to fully comply with the legal requirements for transporting radioactive material, to prosecution with punitive levels of financial penalty and, ultimately, imprisonment.

5.4. Encouraging interaction between stakeholders to develop and share best practices

Competent authorities should be proactive in their approach to assessing the competencies of organizations and the training provided, so as to encourage a regime of compliance and continuous improvement within the industry. In addition, there is a continuing need for competent authorities to interact with one another and to cooperate on matters such as regulatory review and the compliance assessment of package designs.

Links between industry and academia should be encouraged. The associated benefits are:

(a) Providing a cost effective model for industry to conduct both long and short term research focused on specific areas;
(b) Enhancing the technical and scientific standing of the industry amongst the academic community;
(c) Providing a mechanism to attract highly qualified graduates into the industry;
(d) Providing a means to apply current academic methods to the needs of industry.

Links between organizations should be encouraged to promote and enhance common areas such as training, safety assessment techniques and compliance assurance.

6. SPENT FUEL — LIFE CYCLE PROCESSES

The life cycle processes of spent fuel in the context of transportation are shown in Fig. 2. Following its final unloading from the reactor core the spent fuel assembly may be stored in the spent fuel pond, stored in a dry spent fuel store or transferred directly/subsequently to a storage and/or transport cask.

Of importance here are the timescales between unloading the spent fuel from the reactor core and its subsequent transport in the public domain. It is clear that in almost all cases the spent fuel will be transported after decades of

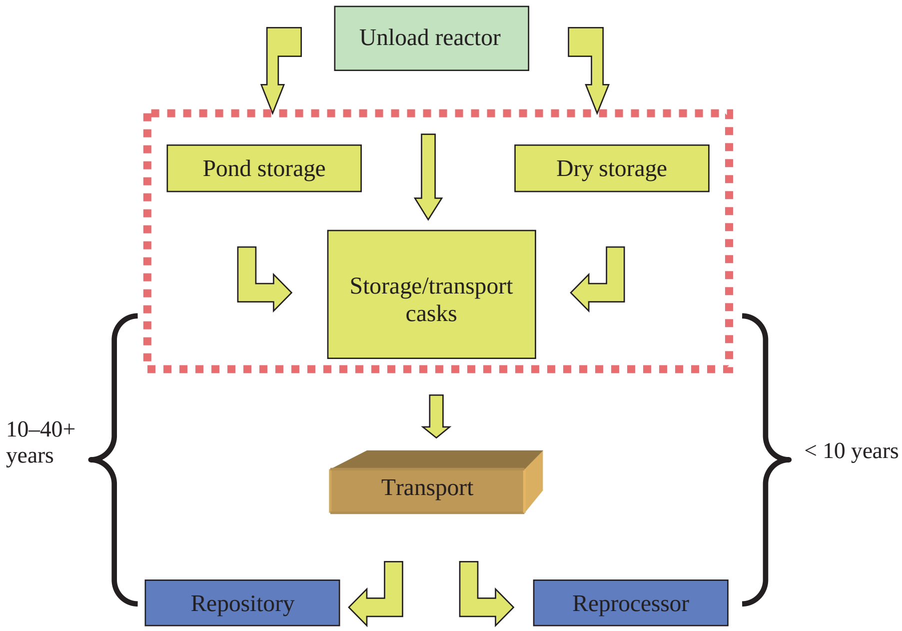

FIG. 2. Life cycle processes of spent fuel in the context of transportation.

storage at the reactor site and that usually transport issues will arise from this spent fuel management strategy.

7. TRANSPORT ISSUES

The transport issues have been grouped under five headings, although the same issue may appear under more than one. The headings follow the logic process of:

(a) Fuel design and utilization;
(b) Spent fuel management strategies;
(c) Maturing nuclear power programmes (worldwide);
(d) Compliance assurance issues;
(e) Competent authority approval processes.

7.1. Fuel design and utilization

The structural integrity of a spent fuel assembly during transport, which includes regulatory accident conditions, is often a vital component of the transport safety case for a spent fuel cask. As the burnup levels of spent fuel increase and the storage time extends for years and possibly decades, it is important that the resistance to deformation of the spent fuel assembly and the resistance to rupture of the fuel pins under prescribed normal transport and accident conditions are demonstrated in the transport cask design safety case.

7.1.1. Development of new fuel assembly designs

To provide performance criteria for the demonstration of the structural integrity of a spent fuel assembly under transport cask accident conditions the following suggestions are made:

(a) Incorporate into the structural design of the fuel assembly the requirements of the transport regulations in terms of the transport package performance impact testing regime;
(b) Give consideration to the physical testing of key structural components to simulate the effects of irradiation and long term storage;
(c) Develop surrogate materials for pseudo spent fuel testing.

7.1.2. Increasing reactor cycle times and fuel burnup

Increasing reactor cycle times and spent fuel burnup levels have a direct effect on the radiation dose profiles of the fuel and the criticality safety case of the spent fuel transport cask. It is therefore necessary to:

(a) Determine the consequential effects of the spent fuel irradiation history upon the structural integrity of a spent fuel assembly. Such effects would include aspects such as pellet cladding interaction, irradiation effects, the material properties of cladding tubes, welded joints, spacer grids and end fittings.
(b) Establish the consequential effects upon the confinement system. For example, the distortion of the fuel pin array and the consequential changes to the fuel/moderator ratio (light water reactor fuel being undermoderated by design).
(c) Consider the consequential effects upon the release fractions from spent fuel pins that are ruptured under transport accident conditions (impact).

(d) Ensure appropriate validation of data libraries and the relative accuracy of criticality analysis codes for various burnup levels when burnup credit is used.

7.1.3. Reactor operating considerations

The fuel management systems used to operate reactors, in terms of the cycle times and discharge burnup, should be taken into account, for example, to take account of an increase in fuel pin replacement operations during the operational life of a fuel assembly. Evidence should be provided to demonstrate that appropriate records are kept to record the actual fuel pin inventory, and also that the repeated dismantling and rebuilding of the fuel assembly does not affect its structural integrity. Other examples include the consequential gamma ray and neutron abundance of spent fuel and their spectra, which would influence the neutron shielding characteristics of the cask design and the management of fuel pin failures (are failed fuel pins removed or do they remain in place, raising issues of hydriding effects during storage?).

7.2. Spent fuel management strategies

Long term storage of spent fuel for years and perhaps decades in transport/storage casks prior to transport raises potential issues which should be considered in the cask safety case. The cask safety case must be periodically reviewed for storage and transport to demonstrate that technical developments and methods of assessment remain at current standards. Degradation of the cask is a major issue and consideration should be given to:

(a) The containment system (seals and restraining system) to establish the effects of the storage environment (inside the cask and external to the cask);
(b) Cask confinement system (creep, hydriding, corrosion of spent fuel assembly and fuel support structure inside the containment system);
(c) Packaging components (corrosion, radiation effects, etc.);
(d) Impact limiters (attachment and performance);
(e) Shielding materials (decomposition, changes in density, creep, etc.).

The effects of the storage environment should also be considered and any changes should include acceptance by those responsible for the spent fuel cask transport safety case.

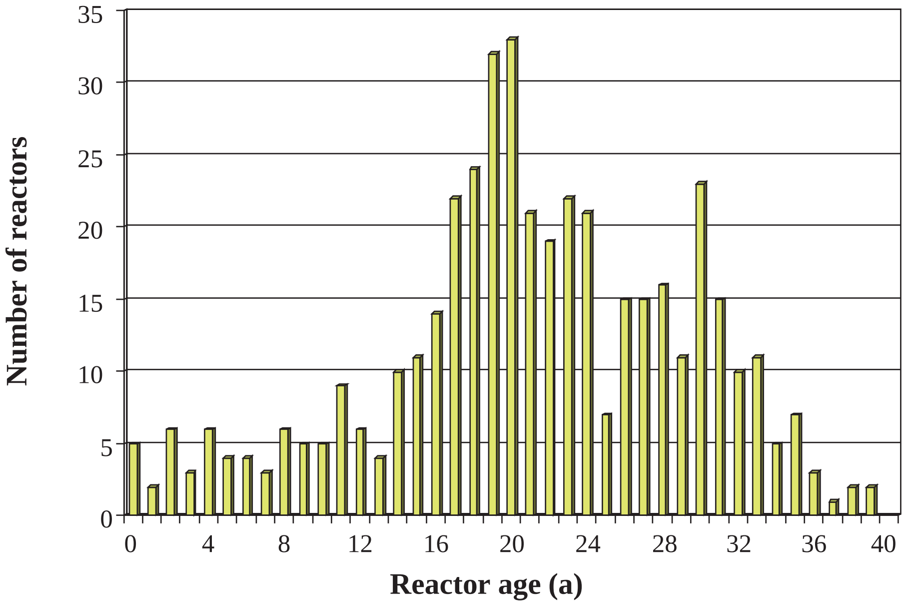

FIG. 3. Age profile of the reactors currently operating throughout the world [1].

7.3. Maturing nuclear power programmes

From Ref. [1] the age profile of the reactors currently operating throughout the world is shown in Fig. 3. From Fig. 3, it can be seen that in ten years time there will be a significant number of reactors that will be over 30 years old and the spent fuel stored on those sites will become a candidate for transport either to buffer stores, reprocessing plants or repositories. Strategies to ensure the future availability of suitably qualified and experienced staff for spent fuel transport should be developed. This is considered necessary as it is expected that increased competition for staff from other parts of the nuclear industry will occur over the next 10–15 years.

The closure of nuclear power plants and the transport of their spent fuel will require close relationships to be encouraged between:

(a) Industry to define operating and disposal strategies at national levels;
(b) Competent authorities to establish the transport safety issues and information needs;
(c) Technical experts/academia to develop the necessary research and development strategies.

7.4. Compliance assurance issues

The long term storage of spent fuel at the nuclear power plant poses several compliance assurance issues as follows:

(a) *Retention of records* — The retention of records for perhaps 30–40 years must be carefully managed, particularly if the spent fuel continues to be stored at the nuclear power plant after the reactor has shut down. Ownership and responsibility for the safe retention of records must be established to ensure their availability at the time of transport.
(b) *Inspection and surveillance regime* — Appropriate inspection and surveillance regimes must be in place throughout the storage period. Inspection and acceptance criteria should be in accordance with the cask design safety case and auditable records should be retained for the storage period of the spent fuel. All records should be made available to the consignor at the time of transport and issued to the disposal site for their retention.
(c) *Control of storage environment* — Appropriate records of the storage environment should be retained to demonstrate that the environment has met the design requirements for the cask design safety case. All measured data should be retained for the period of storage of the spent fuel and all records should be made available to the consignor at the time of the transport and issued to the disposal site for their retention.
(d) *Conventional safety issues* — Features such as those for the handling and 'tie down' of the spent fuel cask should be periodically inspected against prescribed acceptance criteria cited in the cask design safety case.
(e) *Nuclear safety issues* — Where appropriate, nuclear safety issues (for example, any degradation of spent fuel support structure/spent fuel/neutron shielding) should be appropriately monitored.
(f) *Periodic review of cask safety case for transport* — The periodic review process for the cask safety case should, in addition to the cask storage safety case, specifically include the transport safety case to ensure it remains valid and reflects current transport regulatory requirements, technical knowledge and assessment techniques.
(g) *Retention of 'design authority' for the package design* — The 'design authority' of the spent fuel cask design should be clearly identified and all management processes should reflect the responsibility, authority and accountabilities of the design authority in all aspects of the spent fuel cask design.

7.5. Competent authority approval processes

Due to the lengthy timescales often necessary to obtain competent authority (CA) approval and subsequent validation(s) of a spent fuel package design, industry operators endeavour, whenever possible, to submit their applications simultaneously to more than one CA in an effort to reduce assessment time.

To ensure that more efficient use of resources is made by CAs in the approval and validation process, particularly for points of clarification of the cask safety case, there is a need to encourage closer liaison between CAs on individual submissions. This need is further emphasized by the facts that CA resources are finite and cask design safety cases are continually increasing in size and complexity.

There is a growing recognition that harmonization of the structure of cask safety cases and CA guidance material for applicants would be of benefit to the industry and would contribute to a more efficient and effective CA approval and validation process.

8. CONCLUDING REMARKS

The exemplary safety record of spent fuel transport is due to the high professionalism and technical competencies of those involved. The involvement of all stakeholders in the continued development of the IAEA's international regulations and the international agreements in which the requirements feature is a key component in fostering the existing effective safety culture.

Developments in fuel designs to achieve higher levels of burnup and/or longer reactor cycle times, and in fuel management strategies, which result in the storage of spent fuel for decades before shipment, provide significant challenges to the development of transport safety cases.

Linkages between stakeholders need to be encouraged to facilitate the continual improvement and development of spent fuel transport casks and their safety cases to reflect the development of new fuel designs for higher burnup and longer reactor cycle times and the storage and disposal strategies of the future.

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DISCUSSION

W. BRACH (United States of America): I was pleased to see in your presentation an identification of the linkage of transport with storage of spent fuel. The issues you raised are critical both for storage and for transport — whether it be criticality, burnup credit or fuel degradation. I was pleased to see the identification of that linkage.

Also, you stressed the importance of competent authority coordination, and I would note that in the transport community the coordination among the competent authorities around the world is very successful. One of your overheads pointed to the need for better coordination among competent authorities in the review and approval of transport packages, and I would note that there are some countries that have been involved in such collaborative efforts, for example the United Kingdom and France, and Canada and the USA.

You stressed the importance of public outreach and engagement with regard to standards development. Within the IAEA some of the IAEA Safety Standards benefit from the involvement of the public of some countries in providing comments on the publications as they are being developed. Do you see this as an aspect that might be expanded in relation to other IAEA standards and guides?

S. WHITTINGHAM (United Kingdom): We in the United Kingdom are heavily engaged in public acceptance activities. You have to realize that you will never convince everyone. You also have to be aware that sometimes you must use different language. We need to understand what people's concerns are. We have been talking here about regional stores. Clearly, from an industry point of view, if we had a store in Central Europe, one in North America, one in South America and so on, it would make industrial sense. In 50–100 years' time that may happen, but it will not happen if we do not educate the public about what we are trying to do.

E. AMARAL (IAEA): You said at the beginning of your presentation that the Transport Regulations state what should be done, but not who should do it. Later, you said that it is important that the responsibilities of competent authorities and stakeholders be defined somewhere. As you know, the Basic Safety Standards (the BSS) are being revised, and it was agreed at the last meeting of the Commission on Safety Standards that the BSS should be maintained as a comprehensive document linking the Safety Fundamentals with the thematic requirements. Do you think that, if the responsibilities of competent authorities for storage, transport and disposal were addressed in the BSS, this would facilitate the transport issue?

S. WHITTINGHAM (United Kingdom): Certainly, any document that sets out a framework of responsibilities would be very useful.

A. GONZÁLEZ (Argentina — Chairperson): I would like to mention two things which Mr. Whittingham did not mention and which I consider important. One is the need for the international coordination of research and development for resolving particular issues. Transport is a good example of an area where there has been international coordination — in the development of transport casks, in the conduct of safety assessment, and in the investigation of the surface contamination of spent fuel containers.

The second thing I would mention is the issue of international reviews or, as they are called within the IAEA, 'appraisals'. Appraisal is a word used by diplomats instead of the word 'inspection'', which they do not like. Transport is also a good example of an area where appraisals are carried out, and this is an issue I would like to see discussed later in the conference.

CRITICALITY SAFETY

(Session 2)

Chairperson

J. WHANG
Republic of Korea

CURRENT ISSUES IN CRITICALITY SAFETY, INCLUDING BURNUP CREDIT

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Abstract

The paper provides information about state of the art burnup credit criticality safety analysis methodologies and an overview on the national practices, ongoing activities and the regulatory status of the use of burnup credit in different countries. Attention is mainly focused on the principles of choosing calculational procedures, verification and validation of the applied calculational procedures, the impact of conservative assumptions on the predicted isotopic inventory, the reactivity bias due to the isotopic bias, the results of sensitivity and uncertainty studies on the representativeness of experiments with respect to the burnup credit application of interest, and the estimated reactivity effect of burnup profiles.

1. INTRODUCTION

In this paper, attention is mainly focused on the application of burnup credit (BUC) to criticality safety analysis of spent fuel management systems. Application of BUC makes use of the change in the fuel's isotopic composition and hence in its reactivity due to the irradiation of the fuel (Fig. 1). Applying BUC to criticality safety analysis means determining the burnup of the fuel, which results in the maximum neutron multiplication factor allowable for the spent fuel management system of interest, including all mechanical and calculational uncertainties. Accordingly, this burnup is the minimum burnup required for fuel to be loaded in the spent fuel management system. Since the fuel's isotopic composition and hence its reactivity at a given burnup depends on the fuel's initial enrichment (Fig. 1), the minimum required burnup is a function of the initial enrichment. The graph of this function is commonly called the 'loading curve' since it gives, as seen from Fig. 2, the loading criterion for the spent fuel management system of interest. For some systems (as, for instance for the dissolver facility in a reprocessing plant) it is more convenient to present the loading criterion in a different form: instead of indicating the minimum required burnup, the limiting value of a related observable (as for

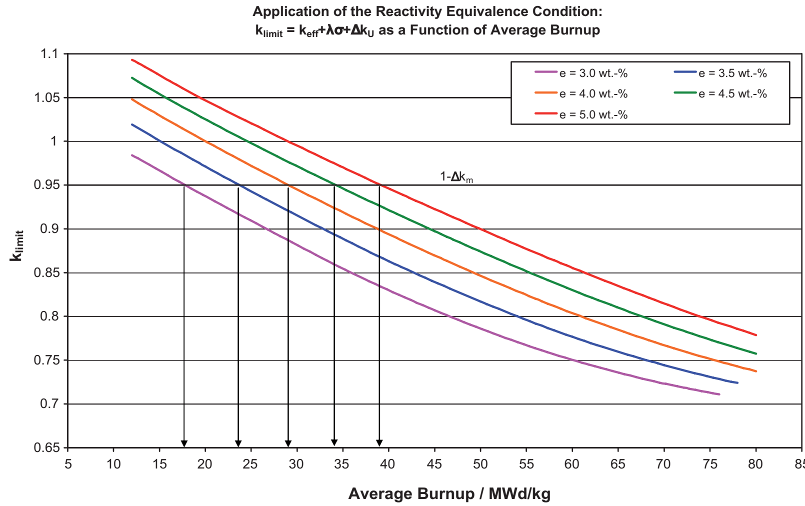


FIG. 1. Example of the neutron multiplication factor of a spent fuel configuration as a function of the average burnup of the spent fuel at different initial enrichments.

instance the maximum allowable fissile content or residual enrichment) is indicated as a function of the initial enrichment.

In whatever form a loading criterion is given the reactivity of a spent fuel management system and hence its loading criterion depends, as always, on the type and design of the fuel to be loaded, on the design of the spent fuel management system, and on the system's condition at which the loading criterion is based (normal operation or an accident condition). The loading curve for a PWR wet storage pond inside a nuclear power plant is usually based on normal operating conditions. For a dry transport cask for spent fuel assemblies, however, it may be necessary to base the loading criterion on an accidental condition (e.g. flooding of the cask with water plus some credible displacement of fuel within the cask).

2. MOTIVATIONS FOR USING BUC

The economic and safety benefits of using BUC have been described on several occasions [1–6]. Sometimes application of BUC is simply required to ensure consistency between the regulatory bases of two spent fuel management systems. An example of such a case has been reported by the United States Nuclear Regulatory Commission (NRC) as set forth below.

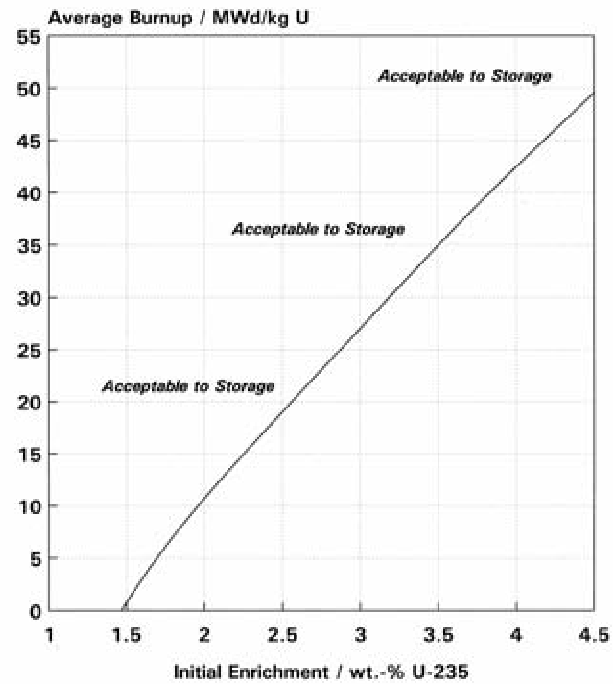


FIG. 2. Example of a BUC loading curve

In Ref. [7], the NRC provides information about potential inconsistencies between the regulatory bases of PWR spent fuel pools and independent spent fuel storage installations (interim storage of spent fuel assemblies in dry cask storage systems). Under Title 10 of the Code of Federal Regulations, Section 50.68 (10 CFR 50.68), the NRC regulates spent fuel pools [8]. Reference [8] allows credit to be taken for a part of the soluble boron present in PWR spent fuel pools to ensure that the neutron multiplication factor k_{eff} of the spent fuel pool inventory does not exceed 0.95 under normal operating conditions. This part is, however, restricted to such an amount that k_{eff} remains less than 1 ($k_{eff} < 1$) under the assumption that the pool is flooded with unborated water. Therefore, by virtue of the double contingency principle the presence of the soluble boron ensures that no individual accident will result in an inadvertent criticality. As explained in Ref. [9], by virtue of the double contingency principle, two unlikely independent and concurrent accidental events are beyond the scope of the required criticality safety analysis. Therefore, if soluble boron is normally present in the spent fuel pool water the loss of soluble boron is regarded as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

The regulatory basis for independent spent fuel storage installations is 10 CFR Part 72 [10]. Under 10 CFR 72.124, the NRC regulates dry cask storage activities to ensure that subcriticality is maintained during handling, packaging, transfer and storage of spent fuel assemblies. It is allowed to take credit for the spent fuel pool soluble boron content to ensure subcriticality during cask loading, unloading and handling operations in the spent fuel pool. However, unlike 10 CFR 50.68, the regulations for criticality prevention in dry storage casks do not place any restrictions on the amount of soluble boron content that may be credited. Accordingly, many cask designs in the USA have incorporated soluble boron credit as a means of increasing dry cask storage capacity without taking credit for the burnup of the fuel, thus simplifying the fuel handling procedure because criticality control does not require a verification of the fuel's burnup. It has been observed that for some cask designs more than 60% of the technical specification soluble boron concentration was credited.

However, since dry storage casks are loaded and unloaded in the cask pit area of a spent fuel pool these casks, while they are in the spent fuel pool, must meet both the 10 CFR 72 and Part 50 requirements. Therefore, these casks, while they are in the spent fuel pool, must remain subcritical without soluble boron credit in order to comply with 10 CFR 50.68. Therefore, as stated in Ref. [7], it is necessary to use BUC in lieu of soluble boron credit.

3. BURNUP CREDIT LEVELS USED

As described in Ref. [11], the change in the fuel's isotopic composition is characterized by the reduction of the concentrations of U-235 (mainly due to fission) and U-238 (mainly due to neutron capture, see Fig. 3), the buildup and burnup of fissile actinides (such as Pu-239 and Pu-241, see Table 1, Fig. 3), the buildup and burnout of actinides acting as neutron absorbers in thermal systems (see Table 1 and Figs 3, 4), the buildup and burnout of fission products (Table 1) and, where applicable, the burnout of integral burnable absorbers initially present in the fuel (either in the form of Gd or Er bearing fuel rods or in the form of IFBA rods containing pellets with burnable absorber coating, usually made of boron). Due to the reactivity worth of the nuclides it is obvious that application of BUC in criticality safety analysis requires consideration of the fissile isotopes, of the reduction of the U-238 content, and of the burnout of integral burnable absorbers (if credited), and allows consideration of any neutron absorbing isotope for which properties and quantities are known with

TABLE 1. ISOTOPES USED IN BUC CRITICALITY ANALYSIS

	Isotope	$T_{1/2}$ [1)]	Cross-sections	Comment
Major actinides	U-235	$7.038 \cdot 10^8$ a	Fig. 3	
	U-238	$4.468 \cdot 10^9$ a	Fig. 3	
	Pu-239	$2.411 \cdot 10^4$ a	Fig. 3	
	Pu-240	6550 a	Fig. 3	
	Pu-241	14.4 a	Fig. 3	
	Pu-242	$3.763 \cdot 10^5$ a	Fig. 3	
	Cm-243	28.5 a		For MOX fuel
	Cm-245	8532 a		For MOX fuel
Minor actinides	U-234	$2.446 \cdot 10^5$ a	Fig. 4	
	U-236	$2.342 \cdot 10^7$ a	Fig. 4	
	Np-237	$2.14 \cdot 10^6$ a	Fig. 4	
	Pu-238	87.74 a	Fig. 4	
	Am-241	432.6 a	Fig. 4	
	Am-243	7370 a	Fig. 4	
	Cm-244	18.11 a		For MOX fuel
	Isotope	$T_{1/2}$ [1)]	$\sigma(n, \gamma)$ [2)]/barn	Comment
Fission products	Mo-95	∞	14.5	
	Tc-99	2.1·105 a	19	
	Ru-101	∞	3.1	
	Rh-103	∞	11 and 135	
	Ag-109	∞	4.5 and 89	
	Cd-113	$9 \cdot 10^{15}$ a	19910	For MOX fuel
	Cs-133	∞	2.5 and 26.5	
	Cs-135	2.10^6 a	8.7	
	Nd-143	∞	325	
	Nd-144	$2.1 \cdot 10^{15}$ a	3.6	For check of Nd-143(n,γ)
	Nd-145	∞	42	
	Nd-146	∞	1.3	For check of Nd-145(n,γ)
	Nd-148	∞	2.48	Burnup indicator
	Nd-150	∞	1.2	Burnup indicator
	Pm-147	2.62a	85 and 96	Decays to Sm-147
	Sm-147	$1.06 \cdot 10^{11}$a	64	

TABLE 1. ISOTOPES USED IN BUC CRITICALITY ANALYSIS (cont.)

Isotope	$T_{1/2}$ 1)	Cross-sections	Comment
Sm-149	∞	41000	
Sm-150	∞	102	
Sm-151	93a	15000	
Sm-152	∞	206	
Eu-153	∞	390	
Eu-155	4.96a	4040	Decays to Gd-155
Gd-155	∞	61000	

1 Half-life; $T_{1/2} = \infty$ denotes a stable nuclide.
2 Thermal cross-section for neutron capture. If two values are given, the first refers to the formation of the product nucleus in the metastable state, the second to the formation in the ground state.

sufficient certainty. Accordingly, the different levels of BUC commonly used are characterized as follows:

(a) Net fissile content level: Credit is taken for the
  — Reduction of the net fissile content due to buildup and burnup of the different fissile nuclides,
  — Reduction of the U-238 content;
(b) Actinide only level: Net fissile content level plus credit for the buildup of neutron absorbing actinides;

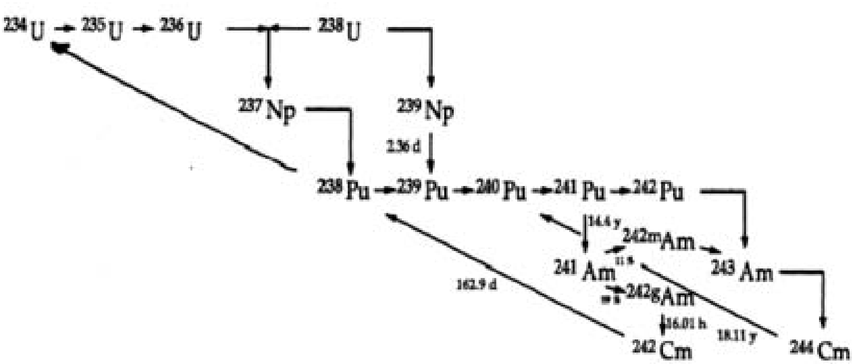
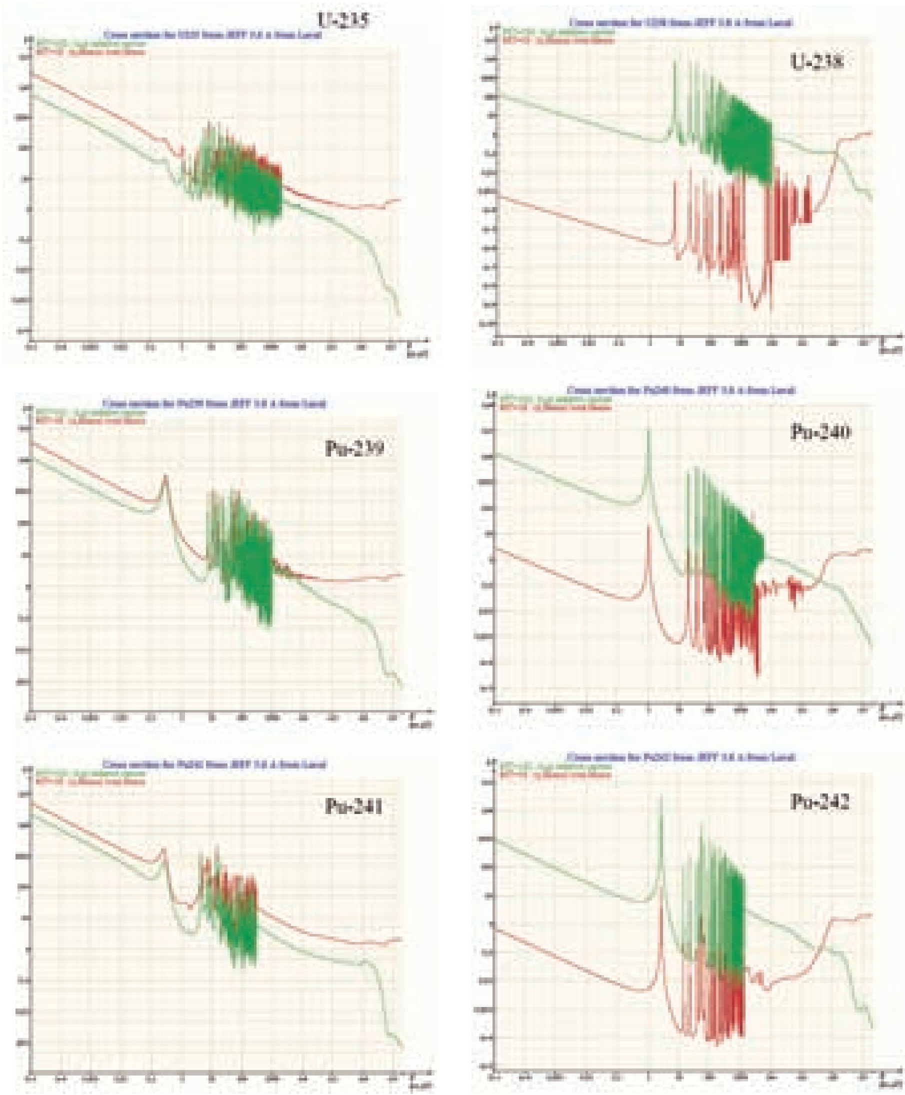


FIG. 3. Buildup and decay of actinides, fission cross-sections (red line) and neutron capture cross-sections (green line) of the major actinides U-235, U-238, Pu-239, Pu-240, Pu-241 and Pu-242 [12]. *(Figure continues on page 153.)*


U-235
U-238
Pu-239
Pu-240
Pu-241
Pu-242

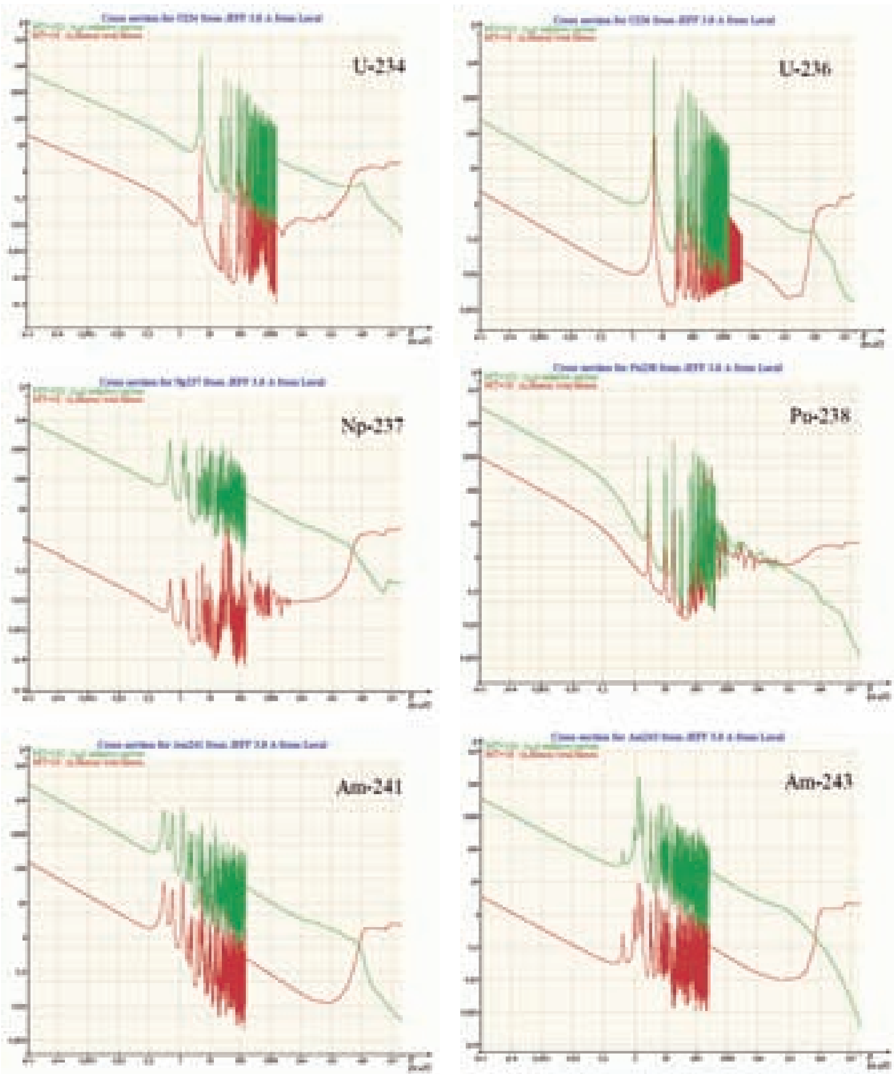


FIG. 4. Buildup and decay of actinides, fission cross-sections (red line) and neutron capture cross-sections (green line) of the minor actinides U-234, U-236, Np-237, Pu-238, Am-241 and Am-243 [12].

(c) Actinide plus fission product level: Actinide only level plus any number of fission products whose use can be verified;

(d) Integral burnable absorber level: One of the BUC levels specified above plus consideration of the presence of integral burnable absorbers in the fuel design.

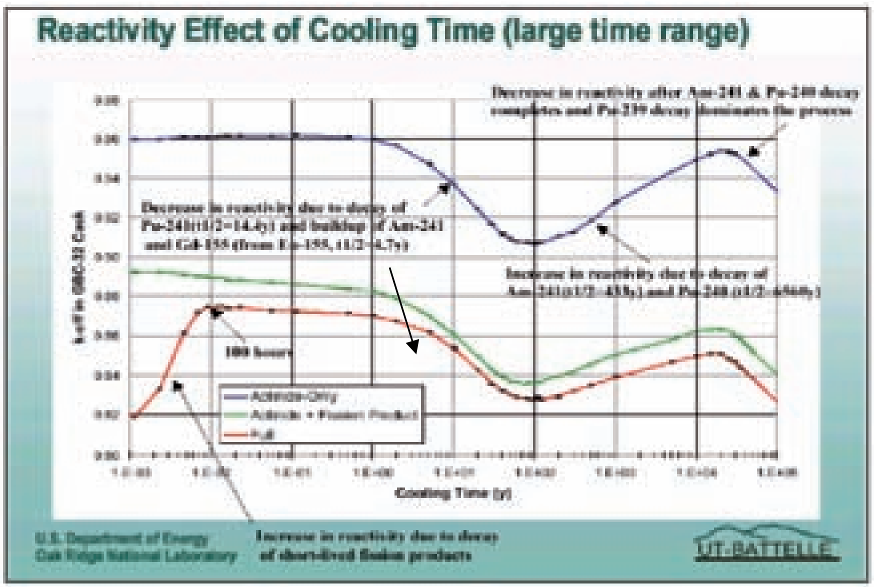


FIG. 5. K_{eff} of a generic BUC cask design as a function of cooling time [13].

Obviously, use of the net fissile content level is a very conservative approach to BUC. Application of the actinide only level still maintains significant conservatism, as is shown in Fig. 5. This figure represents the k_{eff} of a generic transport/storage cask for 32 PWR UOX fuel assemblies [6, 13] as a function of cooling time (time after final discharge from reactor) for a large time range. Application of the actinide only level results in the blue line for k_{eff}, whereas use of the actinide plus fission product level applying the so-called 'principal fission products' Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Eu-155 and Gd-155 (Table 1) leads to the green line in k_{eff}. As shown in Fig. 5, consideration of these principal fission products covers most of the reactivity worth of all the fission products. Except for the first 100 h after discharge, the red line is close to the green line. The red line is obtained for k_{eff} when all the fission products that can be calculated with the depletion code ORIGEN [14] are considered. The increase in the red line during the first 100 hours after discharge is due to the decay of short lived fission products such as Xe-135.

Figure 5 demonstrates the significant change of k_{eff} for long cooling times. Use of BUC can therefore include a cooling time credit for spent fuel management activities such as transport, long term interim storage, reprocessing and final disposal. However, it should be noted that the drop in k_{eff} around its minimum at about 100 years after discharge and the height of the k_{eff}

maximum at approximately 25 000 years after discharge are significantly dependent on the Pu-239, Pu-240 and Pu-241 content of the spent fuel. The change in k_{eff} is therefore significantly dependent on the initial enrichment and the burnup of the fuel and is strongly influenced by neutron spectrum effects during depletion because of the energy dependence of the neutron cross-sections (Fig. 3).

Tables II, III and IV provide national information on practices, ongoing activities and regulatory status in relation to the use of BUC for spent fuel storage and transport in different countries. The information was mainly taken from Ref. [4] and updated by the countries which participated in the IAEA's 2005 Technical Committee Meeting [6].

3.1. Wet storage at reactor

From Table 2 it can be seen that the actinide plus fission product level has been approved and is implemented for PWR UOX fuel wet storage in several countries. However, it should be noted that in some countries, due to regulatory requirements imposed on the validation of the isotopic content of the spent fuel, only the set of principal fission products or subsets of these isotopes are used, whereas in other countries, in the USA in particular, all available fission products except Xe-135 are employed. For wet storage of BWR fuel (UOX as well as MOX) the integral burnable absorber level is usually used. For RBMK fuel there is one reported application in the Russian Federation based on the actinide only level.

3.2. Wet storage away from reactor

Several countries have away from reactor wet storage plants, which are usually unborated. The only away from reactor ponds utilizing burnup credit at the present time are the wet storage ponds of the reprocessing plants in Japan and at La Hague in France. The La Hague pond has approval for actinide only burnup credit for PWR UOX fuel. PWR fuel burnup credit, including some of the principal fission products, is under development. In Sweden, application of burnup credit to the wet storage pond of the CLAB facility is under discussion [6].

3.3. Dry storage of spent fuel

At present, only a few countries are using BUC for dry storage (Table 3). In Armenia, the approved and implemented BUC is restricted to the net fissile content level. In the USA, application of the actinide only level is permitted. In

TABLE 2. USE OF BUC FOR WET STORAGE AT REACTOR (AR)

Country	PWR	BWR	MOX (PWR)	WWER	RBMK	Reactor types[1]
Armenia	n.a.	n.a.	n.a.	INT	Na	WWER
Belgium	APU-1	n.a.	n.a.	n.a.	Na	PWR
Brazil	APU-2	n.a.	n.a.	n.a.	Na	PWR
Bulgaria	n.a.	n.a.	n.a.	INT	Na	WWER
China	INT	n.a.	n.a.	n.a.	Na	PWR
Czech Republic	n.a.	n.a.	n.a.	INT	Na	WWER
Finland	n.a.	Gd	n.a.	INT	Na	WWER, BWR
France	Nc	n.a.	Nc	n.a.	Na	PWR
Germany	APU-2	Gd	APC-2	Nc	Na	PWR, BWR, WWER
Hungary	n.a.	n.a.	n.a.	INT	Na	WWER
Japan	INT	INT	INT	n.a.	Na	PWR, BWR
Korea, Rep. of	APU-2	n.a.	n.a.	n.a.	Na	PWR
Lithuania	n.a.	n.a.	n.a.	n.a.	Nc	RBMK
Mexico	n.a.	Gd	n.a.	n.a.	Na	BWR
Netherlands	APU-2	n.a.	n.a.	n.a.	Na	PWR
Russian Federation	n.a.	n.a.	n.a.	INT	APU-1	WWER, RBMK
Slovakia	n.a.	n.a.	n.a.	UD-2	Na	WWER
Slovenia	APU-2	n.a.	n.a.	n.a.	Na	PWR
South Africa	APU-2	n.a.	n.a.	n.a.	Na	PWR
Spain	APU-2	Gd	n.a.	n.a.	Na	PWR, BWR
Sweden	Nc	Gd	n.a.	n.a.	Na	PWR, BWR
Switzerland	APU-2	Gd	Nc	n.a.	Na	PWR, BWR
Ukraine	n.a.	n.a.	n.a.	INT[2]	No	WWER, RBMK
UK	UD-1	n.a.	n.a.	n.a.	Na	PWR
USA	APU-2	Gd	UD-2	n.a.	Na	PWR, BWR

Abbreviations for Tables 2 and 3:

APU-0/1/2: Approved and implemented BUC using the net fissile content level/actinide only level/ actinide plus fission product level.

APC-1/2: Approved in concept BUC using the actinide only level/actinide plus fission product level.

RR-2: Under regulatory review for taking BUC using the actinide plus fission product level.

UD-1/2: Preparing documentation for taking BUC using the actinide only level/actinide plus fission product level.

Gd: Use of the integral burnable absorber BUC level.

INT: Interested, including some early analysis.

Nc: Not being considered but potentially applicable.

n.a.: Not applicable.

No: No interest since the reactor is shut down.

TABLE 2. USE OF BUC FOR WET STORAGE AT REACTOR (AR) (cont.)

APC-1/2:	Approved in concept BUC using the actinide only level/actinide plus fission product level.
RR-2:	Under regulatory review for taking BUC using the actinide plus fission product level.
UD-1/2:	Preparing documentation for taking BUC using the actinide only level/actinide plus fission product level.
Gd:	Use of the integral burnable absorber BUC level.
INT:	Interested, including some early analysis.
Nc:	Not being considered but potentially applicable.
n.a.:	Not applicable
No:	No interest since the reactor is shut down.

[1] BUC is not currently envisioned for heavy water or gas cooled reactors, so they are not listed.
[2] BUC is allowed by the regulations, but implementation actions have not started.

TABLE 3. USE OF BUC FOR DRY STORAGE

Country	PWR	BWR	MOX (PWR)	WWER	RBMK	Reactor type[1]
Armenia	n.a.	n.a.	n.a.	APU-0	Na	WWER
Belgium	Nc	n.a.	n.a.	n.a.	Na	PWR
Brazil	Nc	n.a.	n.a.	n.a.	Na	PWR
Bulgaria	n.a.	n.a.	n.a.	INT	Na	WWER
China	INT	n.a.	n.a.	n.a.	Na	PWR
Czech Republic	n.a.	n.a.	n.a.	RR-2	Na	WWER
Finland	n.a.	nc	n.a.	nc	Na	WWER, BWR
France	Nc	n.a.	Nc	n.a.	Na	PWR
Germany	APU-1, APC-2	Gd	APC-2	Nc	Na	PWR, BWR, WWER
Hungary	n.a.	n.a.	n.a.	INT	Na	WWER
Japan	Nc	Nc	Nc	n.a.	Na	PWR, BWR
Korea, Rep. of	INT	n.a.	n.a.	n.a.	Na	PWR

TABLE 3. USE OF BUC FOR DRY STORAGE (cont.)

Country	PWR	BWR	MOX (PWR)	WWER	RBMK	Reactor type[1]
Lithuania	n.a.	n.a.	n.a.	n.a.	INT	RBMK
Mexico	n.a.	Nc	n.a.	n.a.	Na	BWR
Netherlands	Nc	n.a.	n.a.	n.a.	Na	PWR
Russian Federation	n.a.	n.a.	n.a.	Nc	INT	WWER, RBMK
Slovakia	n.a.	n.a.	n.a.	UD-2	Na	WWER
Slovenia	Nc	n.a.	n.a.	n.a.	Na	PWR
South Africa	Nc	n.a.	n.a.	n.a.	Na	PWR
Spain	INT	INT	n.a.	n.a.	Na	PWR, BWR
Switzerland	INT	INT	INT	n.a.	Na	PWR, BWR
Ukraine	n.a.	n.a.	n.a.	APU-1	INT[2]	WWER, RBMK
USA	APC-1	INT	INT	n.a.	Na	PWR, BWR

[1] BUC is not currently envisioned for heavy water or gas cooled reactors so they are not listed.
[2] BUC is allowed by the regulatory law but there have been no actions to implement it.

Abbreviations: see Table 2.

Germany and Ukraine this BUC level is already approved and implemented. In Germany, application of the actinide plus fission product level is also allowed. In Slovakia, work is under way to employ this BUC level.

It should be noted that in the USA the following risk informed approach has been discussed. For storage only casks, criticality safety has to be ensured during cask loading (see Section 2.), but it is not an issue once the dry storage cask is on the storage pad since the probability of events leading to inadvertent re-flooding of the cask cavity is considered to be very low. In contrast to this approach, in Germany as well as in Ukraine, reflooding of the cask cavity has to be considered as a design basis event as laid down in the relevant regulations.

3.4. Transport of spent fuel

From Table 4 it can be seen that many countries are using the actinide only BUC level for the transport of spent fuel at the present time. Activities are ongoing in some countries to obtain approval for applying the actinide plus fission product level to transport casks. In Germany, use of this BUC level is already permitted.

TABLE 4. USE OF BUC FOR TRANSPORT (TRANSPORT CASKS)

Country	PWR	BWR	MOX (PWR)	WWER	RBMK	Reactor type[1]
Armenia	n.a.	n.a.	n.a.	APU-0	Na	WWER
Belgium	INT	n.a.	n.a.	n.a.	Na	PWR
Brazil	Nc	n.a.	n.a.	n.a.	Na	PWR
Bulgaria	n.a.	n.a.	n.a.	INT	Na	WWER
China	INT	n.a.	n.a.	n.a.	Na	PWR
Czech Republic	n.a.	n.a.	n.a.	RR-2	Na	WWER
Finland	n.a.	INT	n.a.	INT	Na	WWER, BWR
France	APU-1,UD-2	Nc	UD-1,2	n.a.	Na	PWR
Germany	APU-1, APC-2	Gd	APC-2	Nc	Na	PWR, BWR, WWER
Hungary	n.a.	n.a.	n.a.	INT	Na	WWER
Japan	INT	INT	INT	n.a.	Na	PWR, BWR
Korea, Rep. of	INT	n.a.	n.a.	n.a.	Na	PWR
Lithuania	n.a.	n.a.	n.a.	n.a.	INT	RBMK
Mexico	n.a.	Nc	n.a.	n.a.	Na	BWR
Netherlands	APU-1	n.a.	n.a.	n.a.	Na	PWR
Russian Federation	n.a.	n.a.	n.a.	APU-1	INT	WWER, RBMK
Slovakia	n.a.	n.a.	n.a.	UD-2	Na	WWER
Slovenia	Nc	n.a.	n.a.	n.a.	Na	PWR
South Africa	Nc	n.a.	n.a.	n.a.	Na	PWR
Spain	INT	INT	n.a.	n.a.	Na	PWR, BWR
Sweden	Nc	Nc	n.a.	n.a.	Na	PWR, BWR
Switzerland	APU-1	INT	INT	n.a.	Na	PWR, BWR
Ukraine	n.a.	n.a.	n.a.	INT[2]	INT[2]	WWER, RBMK
UK	INT	Nc	Nc	n.a.	Na	PWR
USA	APC-1, UD-2	INT	INT	n.a.	Na	PWR, BWR

[1] BUC is not currently envisioned for heavy water or gas cooled reactors, so they are not listed.

[2] BUC is allowed by the regulatory law, but there have been no actions to implement it.

Abbreviations: see Table 2.

3.5. Reprocessing

At the La Hague reprocessing plant in France the actinide only BUC level is not only applied to the wet storage pond (see Section 3.2.), but also to the dissolver. For liquids in tanks some specific authorizations with fission-products were obtained. Activities are ongoing to obtain approval for BUC utilizing actinides and at least five of the principal fission products. The actinide only BUC level is also used for the dissolver at the reprocessing plant in the UK. BUC is also used at the reprocessing facilities in the Russian Federation.

3.6. Disposal

The Czech Republic, Germany and the Republic of Korea have performed analyses of the use of BUC in disposal. The USA and Sweden have actively pursued BUC for disposal to cover failed containers containing moderated fuel. The concept of risk informed disposal criticality analysis methodology [15] has been approved by the NRC for use in the Yucca Mountain project, and includes applications of the actinide plus fission product BUC level.

4. DETERMINATION OF A BUC LOADING CURVE

As already discussed in Section 1, the objective of a BUC criticality safety analysis of a spent fuel system is to determine the loading criterion which indicates the minimum burnup (or the limiting value of a related observable) necessary (or maximum allowable) for fuel with a specific initial enrichment to be loaded in the spent fuel management system of interest (e.g. Fig. 2). As illustrated in Fig. 1, the loading criterion is given by the reactivity equivalence condition

$$k_{eff}(e,B,t) + \lambda\sigma(e,B,t) + \Delta k_u(e,B,t) = (1 - \Delta k_m) \tag{1}$$

where the term $(1 - \Delta k_m)$ represents an adequate upper bound of subcriticality. Usually a safety margin of $\Delta k_m = 0.05$ is prescribed as the regulatory basis for a BUC loading criterion.

$k_{eff}(e,B,t)$ is the effective neutron multiplication factor of the spent nuclear fuel system of interest calculated at initial enrichment *e*, average burnup *B* and cooling time *t* of the fuel under the BUC level chosen or permitted by the relevant regulations. $\lambda\sigma(e,B,t)$ represents either the statistical tolerance of $k_{eff}(e,B,t)$ (if a statistical criticality calculation code is used) or the numerical error of $k_{eff}(e,B,t)$ (if a non-statistical criticality calculation procedure is employed). $\Delta k_u(e,B,t)$ in Eq. (1) is the tolerance of the calculated value $k_{eff}(e,B,t)$ which is due to all the biases and uncertainties in the applied BUC calculation route. $\Delta k_u(e,B,t)$ is related to the steps that have to be taken to determine the BUC loading criterion of the spent fuel management system of interest under the BUC level to be applied.

All BUC calculation routes involve the implementation of the following key steps:

(a) Prediction of the isotopic composition of the spent fuel as a function of initial enrichment, burnup and cooling time by means of depletion calculations performed with the aid of a depletion calculation code which is capable of representing, sufficiently accurately, the fuel design characteristics and the reactor operating conditions of interest;
(b) Selection of the BUC isotopes according to the BUC level chosen and validation of the predicted number densities of these isotopes;
(c) Validation of the criticality calculation code intended for performing the criticality evaluations;
(d) Sensitivity studies of reactivity effects of variations and tolerances in the parameters describing the characteristics of the fuel design and the spent fuel managements system of interest under all the normal and accident conditions which have to be considered;
(e) Evaluation of reactivity effects arising from inhomogeneous number density distributions within the fuel (e.g. due to axial and horizontal burnup profiles), taking into account all the normal and accidental conditions that have to be considered in the criticality safety analysis of the spent fuel management system of interest;
(f) Determination of the system's neutron multiplication factor as a function of initial enrichment, burnup and cooling time under the condition of the system on which the loading criterion has to be based (most reactive normal operating condition or a certain abnormal or accidental condition determined by means of the design basis event methodology or a risk informed approach);
(g) Application of the reactivity equivalence condition Eq. (1) to the functions $k_{eff}(e,B,t)$ obtained for k_{eff} under the condition on which the loading criterion has to be based;
(h) If required, analysis of system conditions different from the condition on which the loading criterion is based.

4.1. Isotopic bias and uncertainty

The term $\Delta k_u(e,B,t)$ in Eq. (1) includes the tolerance $\Delta k_{IB}(e,B,t)$ of the spent fuel management system's k_{eff} value arising from the bias and uncertainties in the predicted number densities of the individual isotopes selected according to the BUC level to be applied [11]. Therefore, isotopic validation can be achieved by estimating the tolerance $\Delta k_{IB}(e,B,t)$ from the bias and uncertainty of the concentrations of the selected isotopes obtained from comparison between predicted isotopic concentrations and chemical assay data. A considerable amount of chemical assay data, available in the open literature, have been compiled in the SFCOMPO database by JAERI and OECD/NEA [16]. Table 5 gives an overview of the available chemical assay data. However, as noted in Table 5, part of these data is proprietary to the organizations which participated in the experimental programmes carried out to determine the isotopic composition of spent fuel from commercial reactors. Therefore, the results of these programmes are not generally available.

It is worth noting that, for the first time, a set of assay data from WWER-440 spent fuel, including actinides and fission products, is available outside the Russian Federation [6]. Important steps forward are the finalizations of the REBUS experiments [6] and the PROTEUS experiments. These experiments include actinide and fission product assay data [3] and reactivity worth measurements. The REBUS experiment on the Neckarwestheim II spent fuel can be used for direct validation of calculation tools commonly used in criticality safety analysis. The PROTEUS experiments include several burnup values up to more than 80 MW·d/kg U.

4.2. Criticality validation

The term $\Delta k_u(e,B,t)$ in Eq. (1) includes the tolerance $\Delta k_{CC}(e,B,t)$ resulting from the bias in the calculated k_{eff} value due to

(a) Uncertainties in the employed cross-section data;
(b) Uncertainties arising from differences between available experimental data and the characteristics of the spent fuel management system of interest with respect to the isotopic composition, the geometry of the fuel arrangement and hence the neutron spectrum.

Criticality validation can be achieved by estimating the tolerance $\Delta k_{CC}(e,B,t)$ through evaluation of experimental data which are representative with respect to the spent fuel configuration of interest and the nuclear data characterizing the isotopic compositions and their effect on the spent fuel

TABLE 5. AVAILABLE CHEMICAL ASSAY DATA

Reactor	Fuel type	Remarks
Calvert Cliffs 1	PWR UOX	
H.B. Robinson 2	PWR UOX	
Obrigheim (assay, pellets)	PWR UOX	
Trino Vercelles	PWR UOX	
Turkey Point 3	PWR UOX	
Yankee Rowe, I-V	PWR UOX	
TMI 1	PWR UOX	
H.B. Robinson	PWR UOX	
Takahama-3	PWR UOX	
Goesgen	PWR UOX	ARIANE, proprietary [17]
Beznau	PWR MOX	ARIANE, proprietary [17]
Dodewaard	BWR UOX, MOX	ARIANE, proprietary [17]
Bugey 3	PWR UOX	France, proprietary [17]
Fessenheim 2	PWR UOX	France, proprietary [17]
Gravelines 2 and 3	PWR UOX	France, proprietary [17]
Tihange 1	PWR UOX	France, proprietary [17]
Cruas 4/URT	PWR UOX	France, proprietary [17]
St. Laurent B1 and B2	PWR MOX	France, proprietary [17]
Gravelines 4	PWR MOX	France, proprietary [17]
Gundremmingen B	BWR MOX	proprietary [17]
Neckarwestheim II	PWR UOX	Germany, proprietary [2]
Neckarwestheim II (and other plants)	PWR UOX (and MOX)	REBUS, proprietary [6] and [17]
(different plants)	PWR UOX, MOX, BWR UOX, MOX	PROTEUS, proprietary [17]
Vandellos II	PWR UOX	CSN proprietary,. [18]
Novovoronezh	WWER-440	[6]

reactivity. As reported in Ref. [6], sensitivity/uncertainty evaluation methods have been developed during recent years which allow detailed quantitative comparisons of the similarity of two nuclear fuel systems with respect to the underlying nuclear data characterizing the isotopic compositions in all the material zones of the two systems, their impact on the neutron spectra and hence on the reactivity. These methods can therefore be used to check the representativeness of experimental data and configurations for a spent fuel management system of interest. Available experimental data include Refs [3, 4, 6, 13, 17, 19]:

(1) Critical experiments with fresh fuel in systems similar to spent fuel configurations of interest (validation of cross-sections of actinides and structural and neutron absorbing materials used in the spent fuel management system of interest);
(2) Reactivity worth measurements on individual nuclides and spent fuel compositions (validation of neutron spectrum dependent cross-sections).

An integral check of isotopic compositions, cross-sections and fuel lattice geometry can be performed by calculating reactor critical configurations [13].

4.3. Reactivity effects of axial burnup profiles

The non-uniformity of the axial distribution of the burnup (Fig. 6), and hence the non-uniformity of the axial distribution of the isotopic composition, influences the reactivity of a spent fuel management system. The reactivity effect of an axial burnup profile, often termed the ‘axial end effect’ or ‘end effect’, is usually expressed as the difference $\Delta k_{AX}(e,B,t)$ between the system’s neutron multiplication factor obtained with the axial burnup profile and the system’s neutron multiplication factor obtained by assuming a uniform distribution of the averaged burnup of the profile (Fig. 7).

Whether the reactivity effect $\Delta k_{AX}(e,B,t)$ is positive or not depends on the reactivity importance of the centre region of the active fuel zone relative to the bottom and top end regions of the active fuel zone. The lower the relative reactivity importance of the centre region is, the higher is the reactivity effect $\Delta k_{AX}(e,B,t)$. Because the relative reactivity importance of the centre zone is determined by the axial distribution of the isotopic number densities, the reactivity effect $\Delta k_{AX}(e,B,t)$ is dependent on:

(a) The initial enrichment [4];
(b) The reactor operating conditions (depletion conditions) determining the neutron spectrum in the core and hence, due to the energy dependence of

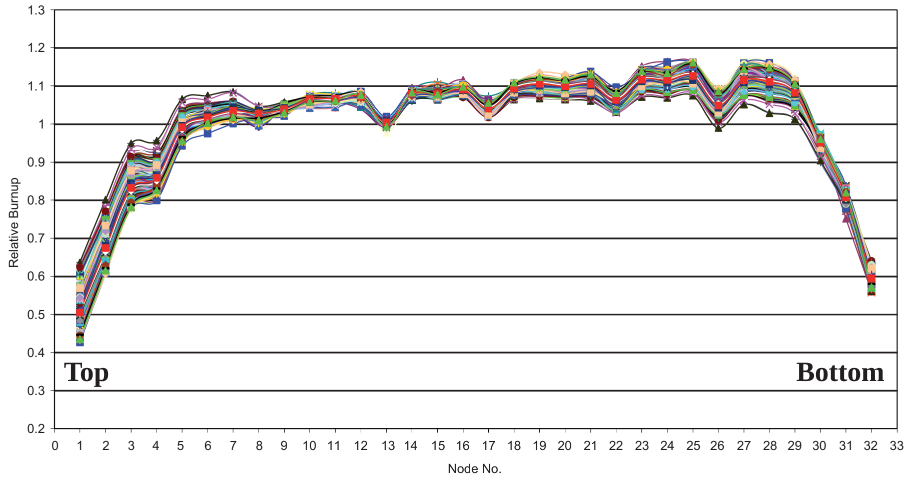


FIG. 6. Example of typical variations in PWR axial burnup profiles (profiles have been normalized by dividing their node specific burnups by the respective average burnup).

the neutron cross-sections (Figs 3 and 4), the change in the isotopic concentrations with increasing burnup [2, 3, 6, 13];

(c) The average burnup [1–4, 6];The cooling time [3].

Axial burnup shapes are mostly asymmetric due to the lower moderator density in the upper half of a core resulting in lower burnup in the top end region of the active fuel zone (Fig. 6). Therefore the reactivity effect, $\Delta k_{AX}(e,B,t)$, is usually determined by the reactivity importance of the centre region relative to the top end region only and is therefore strongly dependent on the asymmetry of an axial burnup profile. In addition, $\Delta k_{AX}(e,B,t)$ is affected by the changes in the neutron spctrum due to axial neutron reflection and absorption conditions and hence the isotopic reactivity worth at the top and bottom end of the active fuel zone.

$\Delta k_{AX}(e,B,t)$ can be included in the term $\Delta k_u(e,B,t)$ of Eq. (1). Another possibility is, as illustrated and explained in Figs 7 and 8, to consider the end effect by generating a correlation between 'equivalent uniform burnup' and average burnup of axial burnup profiles. A third possibility is to use axial burnup profiles directly in the determination of the reactivity equivalence condition (Eq. (1)).

In any case, the question naturally arises of how it can be ensured that the highest positive end effect that has to be considered at a given initial enrichment, given reactor operating conditions, given average burnup, and given cooling time, is bounded by the loading criterion calculated for the spent

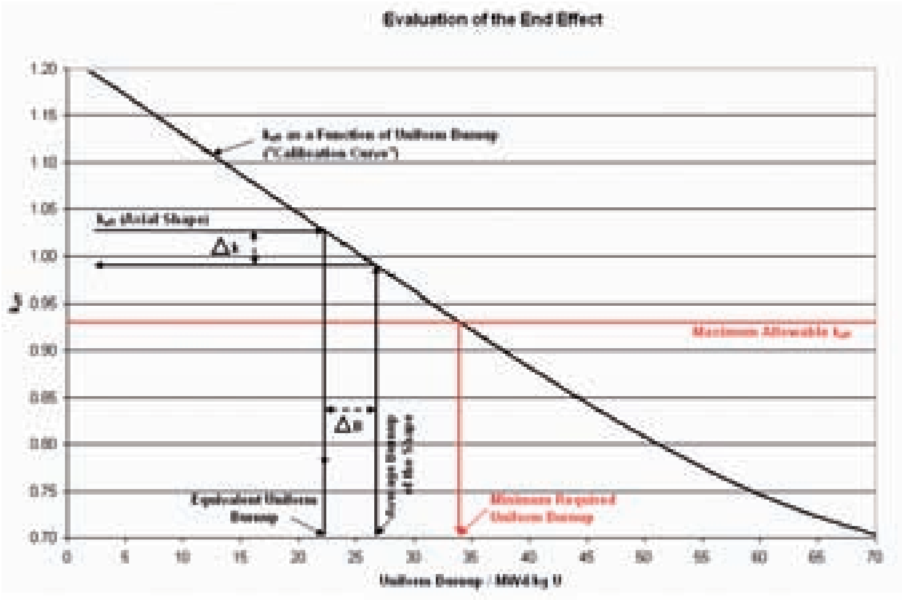


FIG. 7. Variation of the neutron multiplication factor with uniform burnup. The end effect Δk is the difference between the spent nuclear fuel system's neutron multiplication factor obtained with an axial burnup profile (axial shape) and the neutron multiplication factor obtained for the system by assuming the average burnup of the profile uniformly distributed over the active zone of the spent fuel. As illustrated, the end effect can also be expressed by the difference ΔB between average burnup of the profile and the 'equivalent uniform burnup' of the profile given by the neutron multiplication factor obtained with the profile. The end effect can therefore be expressed by a correlation between equivalent uniform burnup and average burnup (Fig. 8).

fuel management system of interest. Therefore, determining a loading criterion implies the need to look for a bounding axial burnup profile that covers the most asymmetric profile occurring in practice (Fig. 6) and hence the highest end effect at a given initial enrichment, given depletion conditions, given average burnup and given cooling time. Since the asymmetry of axial burnup profiles changes with the average burnup, the bounding profile has to be described as a continuous function of the average burnup. A method for generating a bounding profile as a continuous function of the average burnup has been developed in Ref. [21].

4.4. Need for a bounding irradiation history

By definition, a loading criterion specifies a unique average burnup value (or a corresponding value of a related parameter) for a given initial enrichment. A loading criterion accordingly applies to any fuel position of the spent fuel management system of interest and does not take credit for any real loading scheme of the system. A loading criterion must therefore cover not only the variety of axial burnup profiles, as well as horizontal burnup profiles to be considered, but also the variety of irradiation histories of the fuel to be loaded into the system. The task of determining a loading criterion thus implies the need to look for a bounding irradiation history given by those reactor operating conditions (depletion conditions) leading, at given initial enrichment and given burnup, to the highest reactivity of the spent nuclear fuel. As follows from section 4.3, the use of such a bounding irradiation history on the depletion calculations affects the outcomes ($\Delta k_{AX}(e,B,t)$) in relation to the reactivity effects of axial burnup profiles.

The reactor operating conditions (depletion conditions) for PWR, WWER or BWR UOX fuel are characterized by the following parameters:

(a) Specific power and operating history;
(b) Fuel temperature;
(c) Moderator temperature and density;
(d) Presence of soluble boron in the core (PWR);
(e) Core environment (e.g. presence of MOX fuel in the core);
(f) Use of fixed neutron absorbers (control rods or blades, burnable poison rods, axial power shaping rods, control absorber cells);
(g) Use of integral burnable absorbers.

It has already been shown (e.g. in Ref. [3]) how the depletion parameters have to be chosen to ensure application of a bounding history leading, at a given initial enrichment and a given burnup, to the highest reactivity of the spent nuclear fuel. The depletion parameters are related to neutron spectrum hardening. Spectrum hardening results in an increased buildup rate of plutonium due to increased neutron capture in U-238 (Fig. 3) and leads, therefore, to a higher Pu-239 fission rate and hence, at a given power, to a lower U-235 fission rate. Therefore, spectrum hardening has the effect of increasing the reactivity of the spent fuel. Figure 9 illustrates the increase in the concentrations of Pu-239 and U-235 due to spectrum hardening arising from the use of control rods. Since spectrum hardening also increases with increasing burnup,

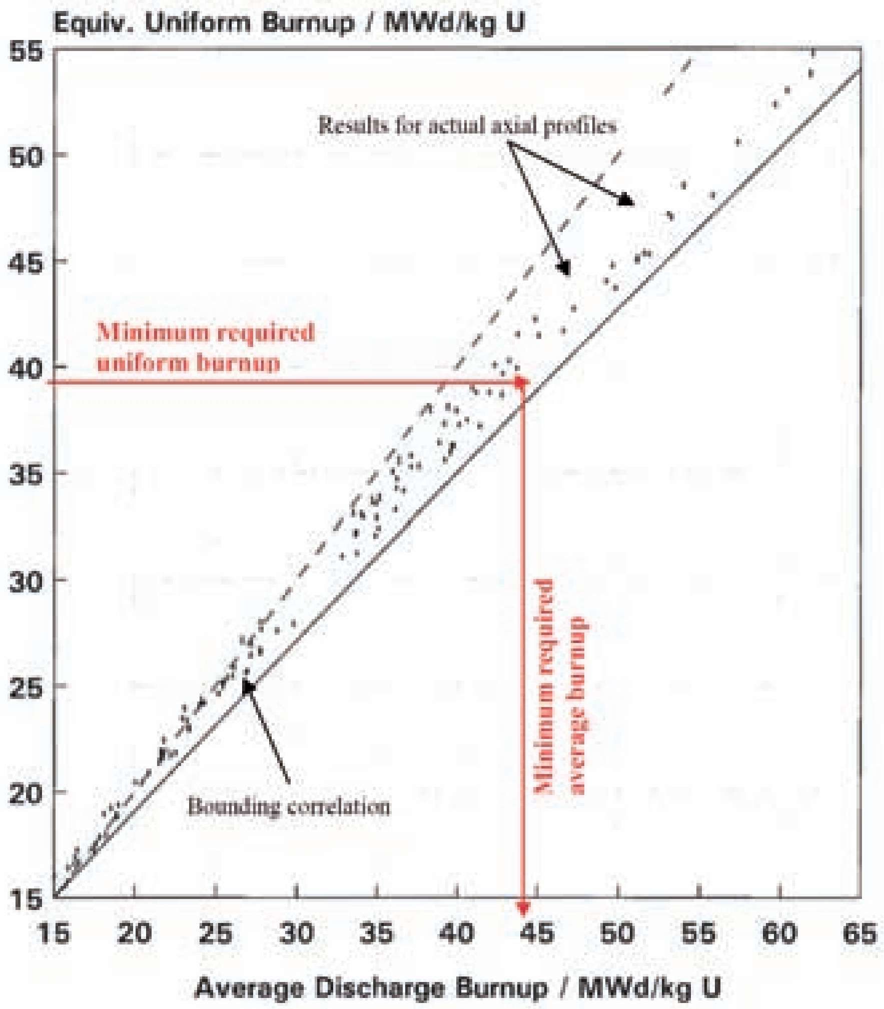


FIG. 8. Correlation between equivalent uniform burnup (Fig. 7) and average burnup: The points shown in this figure represent the results obtained for axial profiles analysed according to Fig. 7. From these results, a bounding correlation between equivalent uniform burnup and average burnup can be derived. This bounding correlation can be used for determining the loading curve at given initial enrichment of the fuel: The intersection of k_{eff} as a function of the uniform burnup with the maximum allowable neutron multiplication factor gives the minimum required uniform burnup (as shown in Fig. 7). This minimum required uniform burnup can be transformed into the minimum required average burnup with the aid of the bounding correlation between equivalent uniform burnup and average burnup (for details, see Ref. [20]).

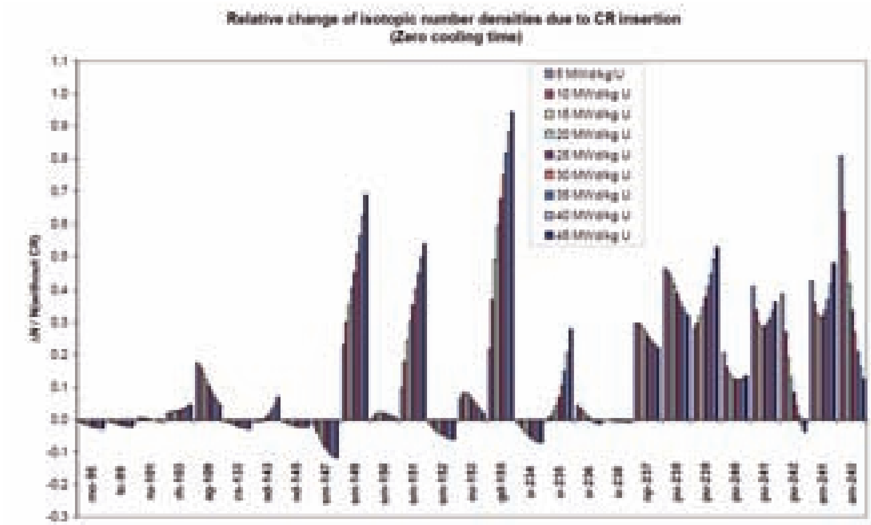


FIG. 9. Relative change of isotopic number densities due to CR insertion in a 17 × 17–(24 + 1) fuel assembly.

Fig. 9 presents the change of the isotopic composition due to control rod (CR) insertion for different periods. The first period ranges from 0 MW·d/kg U to 5 MW·d/kg U, the second from 0 MW·d/kg U to 10 MW·d/kg U, etc., and the last one from 0 MW·d/kg U to 45 MW·d/kg U. As shown in Fig. 9 the relative change in the number densities of U-235 and Pu-239 increases exponentially with an increasing duration of CR insertion. The behaviour of the plutonium isotopes Pu-240, Pu-241 and Pu-242 is dictated by the differences in the energy dependence of the neutron capture cross-sections of Pu-239, Pu-240, Pu-241 and Pu-242, as well as by the energy dependence of the fission cross-sections of Pu-239 and Pu-241 (Fig. 3).

The fission products Sm-149, Sm-151 and Gd-155 are strong neutron absorbers, mainly in the thermal range (Table 1 and Ref. [12]) so that the concentrations of these isotopes in the spent fuel increase with increasing spectrum hardening during depletion. However, the increase in spent fuel's reactivity worth arising from the increase of the concentrations of the fissile actinides dominates by far the decrease in the spent fuel's reactivity worth due to the increase of the concentration of neutron absorbing nuclides. This is demonstrated in Figs 10 and 11 showing the neutron multiplication factor of a generic cask loaded with 21 fuel assemblies of type 17 × 17–(24 + 1) as a function of the CR insertion depth during depletion up to average burnups of 30 MW·d/kg U and 50 MW·d/kg U [22]. From these figures, the increase in k_{eff} with increasing CR insertion depth is significant. The k_{eff} values represented by

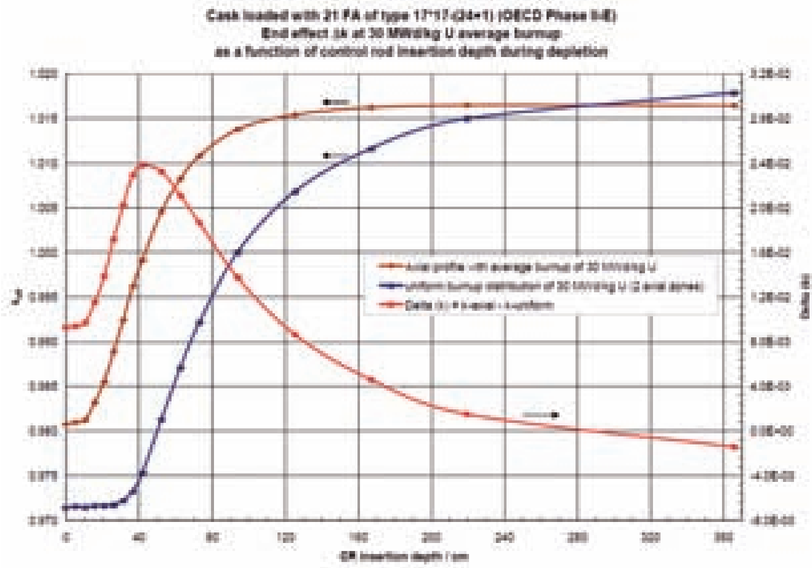


FIG. 10. Neutron multiplication factor, k_{eff} (brown and blue) and axial effects, Δk_{AX} (red) as functions of the CR insertion depth.

the brown lines in these figures refer to the use of a bounding profile generated for 30 MW·d/kg U average burnup and 50 MW·d/kg U average burnup respectively, by means of the procedure described in Ref. [21]. The blue line represents the k_{eff} values obtained by assuming the respective average burnups uniformly distributed over the active fuel zone and dividing this zone into two axial zones according to the CR insertion depth during depletion, in order to account for the change in the isotopic concentrations due to the CR insertion. The red line represents the resulting axial effects Δk_{AX} as a function of the CR insertion depth.

As can be seen, with increasing CR insertion depth, first the relative reactivity importance of the top end region of the fuel zone increases due to the increased fissile content (Fig. 9) and hence the end effect increases. But with further increasing CR insertion depth the relative reactivity importance of the centre region of the fuel zone increases more and more, and finally dominates the increase in the reactivity of the top end zone because of the exponential increase of the U-235 and Pu-239 concentrations with increasing burnup (Fig. 9); the consequence is that the axial end effect Δk_{AX} decreases. For the 30 MW·d/kg U average burnup case (Fig. 10), Δk_{AX} becomes negative at full CR insertion depth, which means that the 'bounding' burnup profile being used no longer represents the bounding case.

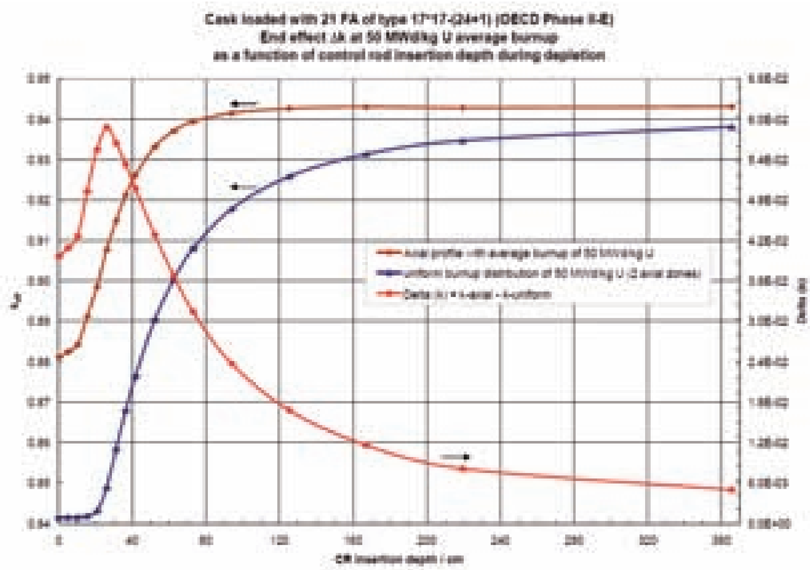


FIG. 11. Neutron multiplication factor, k_{eff} (brown and blue) and axial effects, Δk_{AX} (red) as functions of CR insertion depth.

However, for most of the countries the assumption that CRs are inserted in a fuel assembly during all the irradiation periods of this assembly is also not realistic, nor is the assumption that CR insertion depths greater than 80 cm are used for a total cycle. These assumptions lead to an overly conservative estimate of the neutron spectrum hardening effects due to CR insertion and, hence, to an overly conservative estimate of the k_{eff} value of the system of interest.

Unfortunately, tendencies to choose each of the depletion parameters in such a way that spectrum hardening is maximized have already been observed in practice. In fact, it has been observed for PWR UOX BUC cases that use of the highest fuel temperature and highest moderator temperature (lowest moderator density) has been combined with the use of a very conservatively estimated averaged soluble boron concentration, plus the assumption that the CRs are inserted at least 80 cm during all the irradiation periods, plus the assumption that the UOX fuel assemblies are completely surrounded by MOX of the highest possible plutonium content. Maximizing spectrum hardening in such a way produces an extremely unrealistic evaluation of the situation. It should be kept in mind that the chosen depletion parameter combination has a significant impact on:

(1) The outcome of the validation of the predicted isotopic inventory;
(2) The outcome of the criticality validation performed with the aid of a set of experiments selected by means of sensitivity/uncertainty tools;
(3) The reactivity effects due to variations and tolerances in the parameters describing the spent fuel system of interest;
(4) The outcomes for reactivity effects due to axial and horizontal burnup profiles.

Unnecessary maximizing of spectrum hardening can lead to overestimation of the correction of the predicted isotopic inventory. Chemical assay data usually originate from commercial fuel. Isotopic correction factors are therefore based on comparisons of these assay data to calculated data predicted by using the real irradiation histories of the fuel rods from which the assays were taken. Application of correction factors greater than 1 to a fissile content that is already overestimated results in an overly conservative estimated k_{eff} value. Unnecessary maximizing of spectrum hardening restricts the experimental data and configurations which are accepted as representative for a spent UOX fuel system of interest to MOX fuel data and configurations only [6]. Criticality validation is thus unnecessarily constricted.

Unnecessary maximizing of spectrum hardening can be misleading when the loading criterion of the spent fuel system of interest is directly determined by using an axial burnup profile, which is assumed to be bounding due to its shape but which is no longer bounding due to the overestimation of the relative reactivity importance of the centre region of the active fuel zone. Credit should not be taken for negative axial end effects since this leads, for small average burnups in particular, to wrong conclusions when an axial profile is used which is assumed to be ‘bounding in shape’.

Unnecessary maximizing of spectrum hardening results in a significant reduction of the economic benefit and, hence, of the safety benefit of BUC, since more neutron absorbing material and/or greater distances between the fuel positions inside the spent fuel system of interest are required. A reasonable choice of the set of depletion parameters suitable for the plant(s) of interest should be based on statistics of realistic reactor operation strategies and irradiation histories. As reported in Ref. [6], activities are ongoing to sample such statistics.

4.5. Conclusions for the BUC calculation routes to be used

As follows from the preceding sections the depletion code used in a BUC criticality safety analysis should be capable of representing sufficiently accurately:

(a) The complexity of the fuel assembly design of interest (presence of integral burnable absorbers, horizontally and/or axially inhomogeneous distribution of the initial enrichment);
(b) The effect of the presence of burnable poison or CRs on the isotopic inventory of the spent fuel.

Two dimensional depletion methods are generally preferable to one dimensional depletion calculation routes, in particular when the fuel design of interest has horizontally non-uniform enrichment distributions, when integral burnable absorbers are present in the fuel, or when CRs are inserted during an irradiation period.

At the Technical Committee Meeting in 2005 [6], presentations were made giving information about the ongoing activities in developing and validating three dimensional depletion calculation procedures. Application of three dimensional depletion calculation methods in BUC criticality safety analysis will be of particular interest because of:

(1) The non-uniformity of the axial isotopic composition distribution due to the non-uniformity of axial power and hence burnup distributions;
(2) The capability of three dimensional codes to include the neutron reflection conditions at the top and bottom ends of the fuel zone which affect the axial isotopic composition distribution;
(3) The capability of three dimensional codes to represent axial enrichment zoning;
(4) The capability of three dimensional codes to take account of the fact that neither integral burnable absorbers nor burnable poison rods extend over the full active length;
(5) The capability of three dimensional codes to take account of the fact that CRs are usually not fully inserted during irradiation and that the insertion depth, if CRs are used, is usually significantly less than 80 cm.

The criticality calculation procedure should have the capability to represent:

(i) All the above mentioned complexities of the fuel assembly designs;
(ii) The complex geometries of the fuel management system of interest;
(iii) The axially and horizontally non-uniform burnup distributions.

It is state of the art, therefore, to use three dimensional criticality codes employing Monte Carlo techniques and using adequate cross-section libraries containing the nuclides to be used in BUC.

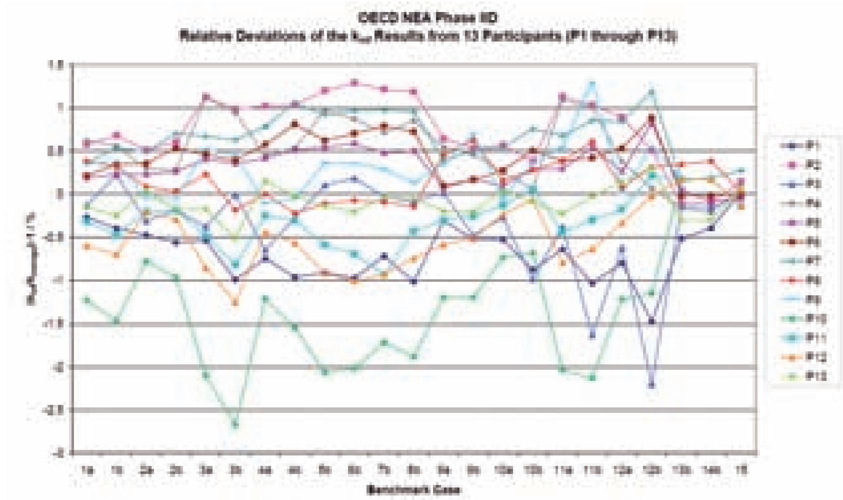


FIG. 12. Example of the relative differences in calculated neutron multiplication factors due to the use of different depletion codes for predicting the isotopic inventory and the use of different criticality calculation codes (from Ref. [23], Table 4.1).

Figure 12 shows results from a benchmark study on the effect of the use of CRs on the isotopic composition of spent fuel and hence on the fuel's reactivity. This benchmark study was conducted by the Expert Group on Burnup Credit Criticality Safety under the auspices of the OECD/NEA [23]. Each contributor to the benchmark used his own depletion calculation procedure (unless otherwise specified in Fig. 12) and his own criticality calculation procedure. Figure 12 shows that, apart from contribution 'P10' (in this contribution a one dimensional depletion calculation procedure was used for comparison) the relative deviations of the neutron multiplication factors calculated by the contributors are within an interval smaller than –1.5%, +1.5%. This demonstrates the high quality reached in BUC criticality safety analysis.

— Benchmark cases 2 and 4 through 8: With CR insertion during depletion, cooling time 0 years;
— Benchmark cases 10 and 12: With CR insertion during depletion, cooling time 5 years;
— Benchmark cases 1 and 3: No CR insertion during depletion, cooling time 0 years;
— Benchmark cases 9 and 11: No CR insertion during depletion, cooling time 5 years;

— Benchmark cases 1a, 2a, 3a, 4a, 9a, 10a, 11a, 12a: Actinide only BUC[1]; all the other cases: Actinide + fission product BUC$^{+)}$;
— Benchmark case 13b: No CR insertion, cooling time 5 years, pre-defined isotopic number densities for all participants;
— Benchmark case 14b: With CR insertion, cooling time 5 years, pre-defined isotopic number densities for all participants;
— Benchmark case 15: Fresh fuel, pre-defined isotopic number densities for all participants.

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[1] Isotopes used (cf. Table I): Actinides: U-234, U-235, U-236, U-238, Np,-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243; Fission products: Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155; Others: O-16.

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DISCUSSION

Y. KOVBASENKO (Ukraine): In your opinion, how many isotopes are really acceptable for burnup credit calculations for PWR spent fuel management systems?

J.-C. NEUBER (Germany): It depends on the spent fuel, the spent fuel system, and the conditions you have to analyse. For instance, for a wet or water-flooded storage system you have to use the actinides uranium-235, uranium-238, plutonium-239 and plutonium-241; and you may use the plutonium isotopes 238, 240 and 242, as well as neptunium-237 and the americium isotopes 241 and 243. You may use a set of about 15 fission products, which is often called the 'set of principal fission products'. One of these fission products is caesium-135. Under normal operating conditions in a wet storage system, or cold conditions in a flooded system, caesium is not volatile. But in the case of a transport/storage cask under fire conditions caesium becomes volatile, so that the distribution of caesium within the fuel becomes unknown. So when reflooding of the cask has to be analysed, caesium isotopes should not be included in the set of isotopes used in the burnup credit criticality safety analysis (usually, i.e. if you have no better information). This is an example. In principle, due to physics, you have to include all the nuclides with positive reactivity worth; and you may include all nuclides with negative reactivity worth — the use of which can be validated. Don't include short lived isotopes because they usually can't be validated.

Y. KOVBASENKO (Ukraine): So, with respect to available validation data, the number of isotopes I can use is approximately 20–30 isotopes — not more.

J.-C. NEUBER (Germany): No, in principle there are more than 30 that can be used. You will find a list of burnup credit isotopes in my paper. It should be mentioned, however, that in the USA for instance, it is allowed in the criticality safety analysis of wet storage systems to use all the fission products that can be calculated, except for xenon-135. However, this is not the European way because we have, according to our relevant standards, in particular, the German standards for the use of burnup credit, to validate the calculated number densities with the aid of experimental data.

T. SAEGUSO (Japan): Could you share with us your thoughts about how to avoid the faulty loading of unknown fuel, such as fresh fuel, into the system?

J.-C. NEUBER (Germany): As in any other accident condition, the double contingency principle is applied to the misloading event. However, there are different ways of application of the double contingency principle in the world and, in addition, the way of application may depend on the spent fuel management system one is talking about.

Concerning borated wet storage pools inside PWR nuclear power plants, for example, the misloading of a fuel assembly which does not comply with the loading curve, i.e. which does not have enough burnup, is considered as one accident condition. By virtue of the double contingency principle a second independent, unlikely and concurrent accident event need not be assumed. Therefore, the boron content of the storage pool may be credited — since a concurrent dilution event need not be assumed. This is the usual approach in the USA for instance, so that one assumes for a PWR wet storage pool either a misloading event or a boron dilution event. It is not necessary to combine the two events.

The German approach is different, because we take into consideration that no system can withstand the misloading of more than one fuel assembly if it cannot withstand the misloading of one fuel assembly. This is due to the fact that a misloading error results from operational errors in the fuel handling procedure or errors in the burnup information from the reactor records. In any case, if a misloading event occurs there is a great probability that this event will remain undetected despite the control checks performed during or at the end of a fuel loading campaign. If the error remains undetected, then, for example a boron dilution event taking place at a later time in a PWR wet storage system cannot be regarded as a concurrent event. The double contingency principle says that one does not need to consider two events which are independent and concurrent and which have a low probability of occurrence. But if the misloading event is not detected, a later boron dilution event is not concurrent. So therefore, according to the double contingency principle, one has to analyse the combination of the misloading event and the later boron dilution event.

At a later time, if you again load fuel into the wet storage system for which you are using burnup credit the fault can be repeated. Then you have two undetected misloading events and you have, therefore, to combine these two misloading events with a later boron dilution event; and so on and so forth. At the end you will come out with all storage positions loaded with fresh fuel; and this scenario has to be combined with a later dilution event. This combination makes it impossible to use burnup credit.

So our conclusion was that we have to apply the double contingency principle directly to the misloading event in such a way that at least two unlikely, independent and concurrent events have to occur before a misloading event can occur. Each step of a fuel handling procedure has to comply with this application of the double contingency principle. This is laid down in our burnup credit criticality safety regulations.

With respect to spent fuel casks, in particular dry storage casks, I have to add that we have to include intentional or inadvertent reflooding of the casks with pure water in our criticality safety analysis. This makes it necessary to rule the misloading event out by applying the double contingency principle to the misloading event directly. In other countries, as for instance in the USA, the situation is different: reflooding need not be considered and, therefore, a risk informed approach is taken consisting of estimating the probability of reflooding a dry storage cask sitting in its pad in the dry storage facility.

So you see, there are significant differences in the philosophy between different countries regarding the misloading event.

POTENTIAL BENEFITS OF BURNUP CREDIT IN HUNGARY

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Abstract

According to the present Hungarian regulations, the criticality safety analysis of a transport/storage device should be based on the fresh fuel assumption. It was recognized that for new advanced fuel types with higher enrichment, some of the existing transport/storage facilities can only be used with reduced capacity unless burnup credit is used in the subcriticality analysis. The impact of burnup credit on the capacity of these facilities, for the compact storage pool and for the TK-6 transport cask is examined. Preliminary investigations have been made on some aspects of the safe application of burnup credit. These are the study of the influence of uncertainty in nuclear data on criticality calculations, the study of the influence of the axial burnup distribution on criticality and the testing of the depletion calculation methodology with newly available post-irradiation measurements.

1. INTRODUCTION

Hungary has four WWER-440 units at the Paks nuclear power plant. These four units provided 38.6% of the total production of electric energy in the country in 2005. The original fuel assembly has a 1.22 cm pitch and 3.6% maximum enrichment. Recently the enrichment was increased to 3.82% with radial zoning, and a change of lattice pitch from 1.22 cm to 1.23 cm is now being considered.

After removal from the core the spent fuel is loaded into the wet storage pool at the reactor. The pool contains two racks. The upper rack has a 22.5 cm spacing between the assemblies; subcriticality is ensured by the water among the assemblies. The lower rack has compact storage; it contains boron steel plates and the assembly spacing is 16 cm. After some years of cooling the spent assemblies are transported either to the dry interim storage module using a C-30 transport cask or to the Russian Federation using a TK-6 transport cask. The maximum storage capacity of both casks is 30 assemblies.

According to Hungarian regulations, the criticality safety analysis of a transport/storage device must be based on the fresh fuel assumption. The effective multiplication factor should be less than 0.95 for all normal conditions and should be less than 0.98 for any single failure case. Meeting these subcriticality requirements should be ensured by a conservative safety margin covering all kinds of uncertainties. This requirement can be written as:

$$k_c + \Delta k_c \leq USL \tag{1}$$

where k_c is the calculated value of the multiplication factor, Δk_c is the error of the calculation and USL is the upper safety limit, which already contains the administrative safety margin (usually 0.5) and all of the uncertainties, including the error in the estimation of cross-sections. This uncertainty should be determined using critical benchmark experiments on assemblies similar to the device being investigated. For fresh fuel it can be derived from UO_2 criticality experiments.

If the full capacity is used, compliance with the subcriticality criteria can be demonstrated by the fresh fuel assumption for the upper rack of the storage pool, for the interim dry storage modules and for the cask used for transport between the pool and the dry storage module. However, using the fresh fuel assumption, the 3.82% fuel in the compact rack and 3.6% fuel in the TK-6 cask under optimal moderation conditions do not meet the subcriticality criteria if they are used to full capacity. At present, subcriticality is ensured by technical measures in these facilities, i.e. they are used with a reduced capacity.

Preliminary investigations of the benefits of using burnup credit for these facilities were carried out. They are described in Sections 2 and 3. For safe application of burnup credit, the uncertainties associated with the cross-section and with the accuracy of composition calculations should be analysed. The possibility of performing these investigations using the available experimental data is investigated in Sections 4 and 5.

2. SUBCRITICALITY OF THE COMPACT STORAGE RACK

The compact storage pool has its maximum multiplication properties with a water density of 1 g/cm^3. In the case of assemblies with 3.82% enrichment and 1.23 cm pitch the subcriticality criteria are met if absorber assemblies are placed in every 25th position. This means a reduction of approximately 4% in storage capacity. This capacity decrease could be avoided by application of burnup credit.

For reliable determination of the required average assembly burnup, the influence of the burnup distribution on the multiplication factor should be taken into account. From previous experience it is known that k_{eff} is not determined unequivocally by the average assembly burnup, but may depend significantly on the axial change of the burnup. This influence is generally referred to as the 'end effect' because of the importance of the top and bottom end of the fuel assembly having lower burnup. This phenomenon may vary from plant to plant, depending on the fuel type, the loading strategies, etc. To account for this influence in this particular case a number of criticality calculations were performed for the compact storage rack, using real axial assembly burnup distributions as well as with uniform burnup distributions corresponding to the average of the real axial burnup distribution. The axial burnup distributions were 'real life' distributions, derived from the KOLA benchmark [1]. This benchmark definition contains the reload patterns and detailed operational histories of the first 12 cycles of unit 3 of the KOLA nuclear power plant. From the fifth cycle the core contained assemblies with 4.4% enrichment. Several assemblies achieved average burnup levels up to 50 MW·d/kg U and a few assemblies even higher levels. The reason for choosing this benchmark as a source for the distribution is that higher enrichment fuel (4.4%) was used and higher burnup was achieved than is the case for the usual WWER-440 unit. Details of the burnup and composition calculations are given in Ref. [2]. The calculations were performed using 'actinides only' and 'actinides + fission products' burnup credit. The actinides U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243 and the fission products Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155 were considered. About 200 axial assembly burnup distributions were investigated. Assemblies with 3.6% and 4.4% enrichment were considered.

Approximately the same maximum end effect was found for 3.6% and 4.4% assemblies. This value was about 3% using fission products + actinides and about 1.5% using actinides only. The influence of the radial burnup distribution was also investigated for 20 assemblies which were close to the core boundary during their last cycle in the core. No statistically significant influence on k_{eff} was found.

Taking into account the effect of the axial distribution it was found that the subcriticality of the compact rack can be ensured if the assembly burnup B meets the requirement that

$B > 11$ MW·d/kg U	using the actinides only approach, and
$B > 8.5$ MW·d/kg U	if the actinides + fission products approach is used.

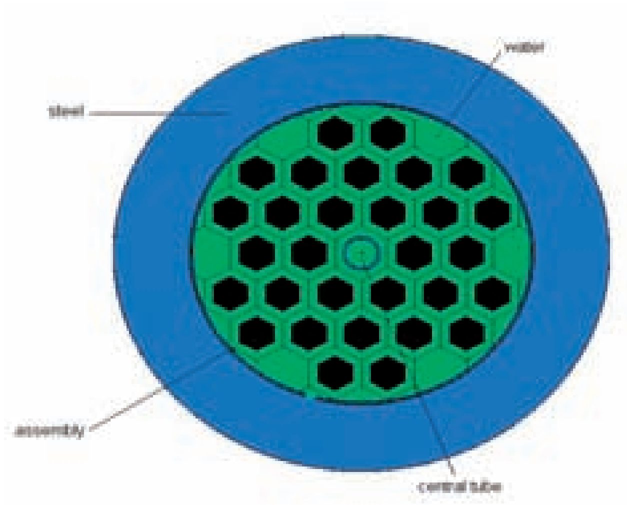

FIG. 1. Horizontal cross-section of the TK-6 transport cask.

It is noted that these values of burnup are generally achieved during one cycle.

3. SUBCRITICALITY OF THE TK-6 TRANSPORT CASK

The TK-6 transport cask is intended to be used for transporting spent fuel assemblies to the Russian Federation. It is a steel container 392 cm in height and with an outer radius of 106 cm. Inside the container there is a basket containing the assemblies in a hexagonal lattice. The lattice pitch of the assembly is 22.5 cm. A maximum of 30 assemblies can be loaded into the cask. During the transport operation, the assemblies are stored under water. Above the water level the cask is filled with nitrogen. The horizontal cross-section of the cask is shown in Fig. 1.

For normal water density this cask is subcritical with 30 assemblies of the type described above. However, there is a water drain valve at the bottom of the cask. An accident that involves the valve being broken cannot be excluded. In this case the water would leak out at the bottom of the closed cask and the water level and the pressure above the water level would decrease. A thermo-hydraulic analysis of this situation is not available, so an investigation of the optimal moderation conditions is necessary. The results of the analysis show that a cask containing 30 fuel assemblies with 3.6% enrichment would not meet the subcriticality criteria. Subcriticality can be ennsured by technical measures,

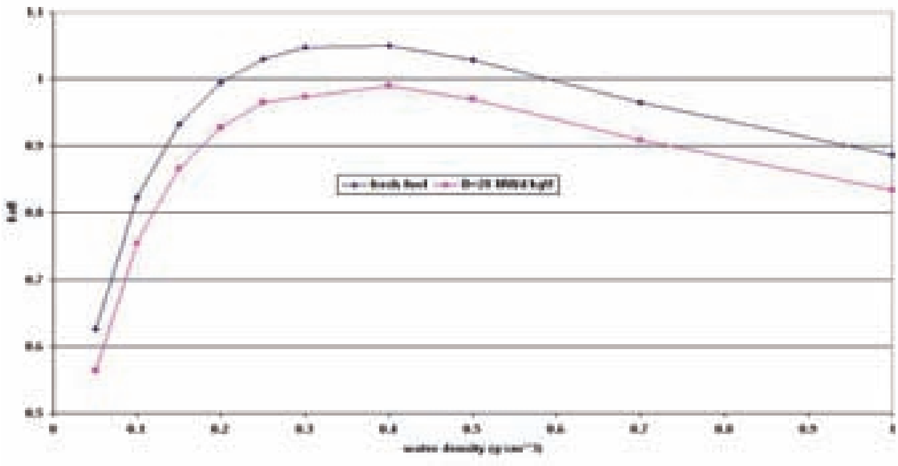

FIG. 2. Multiplication factor of TK-6 as a function of water density with the fresh fuel assumption and with burnup credit.

such as loading only 24 assemblies into the cask or loading 27 assemblies and 3 absorber assemblies. This is a substantial reduction of the transport capacity, which could be avoided by using burnup credit.

Simplified analysis shows that for an assumption of about 28 MW·d/kg U average assembly burnup the cask would meet the subcriticality criteria in the optimal moderation conditions. In this analysis, only the actinides were used and the same extent of end effect was assumed as in the previous case. The calculated multiplication factor of the cask as a function of water density is shown in Fig. 2 with the fresh fuel assumption and with burnup credit.

4. INFLUENCE OF THE PLUTONIUM AND URANIUM CROSS-SECTION UNCERTAINTIES IN BURNUP CREDIT APPLICATION

Errors in cross-section data represent a major source of uncertainty in criticality calculations. This uncertainty should be taken into account using criticality benchmark experiments. A set of criticality experiments should be selected with relevant neutron physical characteristics (enrichment, lattice pitch, hydrogen/uranium ratio, etc.) similar to the investigated application. Criticality calculations should be performed for this set of experiments using the computational tool to be validated (code + data library). Using statistical methods for comparison of the calculated and measured multiplication factors, an upper safety limit (USL) containing the bias and uncertainties associated with that particular computational tool can be derived.

Such a method for the derivation of the USL was developed at the Oak Ridge National Laboratory (ORNL) in the USA. The USLSTAT code for implementing these methods was also developed at ORNL. They are described in detail in Ref. [1]. This method, involving the use of a confidence band with an administrative margin approach, has been used for criticality safety analysis in KFKI AERI in recent years. In its general form the derived USL can be written as

$$USL(x) = 1 - \Delta k_m - W - \beta(x)$$

where x is the physical parameter which gives the most conservative USL in the trending analysis, Δk_m is the administrative safety margin (usually 0.05), $\beta(x)$ is the bias, i.e. the difference between the linear fit and the measured k_{eff} values. W is the confidence bandwidth for the lower confidence limit. For a specified confidence level, W is determined by the deviation of calculated and measured k_{eff} values.

While it is relatively easy to find appropriate benchmark experiments for fresh fuel analysis, it is more difficult for burnup credit applications. There are plenty of UO_2 critical experiments, but there are no publicly available experiments for burned fuel. For criticality, the most important elements are uranium and plutonium. A possible approach therefore is to use MOX criticality experiments for this purpose. However, the question of the extent to which these experiments are adequate for real transport/storage application may arise.

In an attempt to derive an upper subcriticality limit for burnup credit applications, 132 MOX experiments were selected and investigated using the ICSBEP Handbook. Although this is a large number of experiments, their characteristics only partly cover the neutronic features of the applications of interest (compact storage and TK-6 transport cask). The PuO_2 content of the MOX fuel varies from 1.5% up to 20% in these experiments. The lower bound of this interval is close to the plutonium content of the burned fuel of interest in the case being investigated. The ‘plutonium enrichment’, i.e. the ratio of fissionable plutonium atoms to the total number of plutonium atoms, ranges from approximately 70% to 90%, which is applicable for the cases being investigated, where this quantity is about 75–77%. The lattice pitch is larger in all experiments compared to those which exist in WWER-440 fuel, but this could be extrapolated. However, in all experiments, natural or depleted uranium dioxide was used, i.e. the U-235 content is only approximately 0.7 or 0.2%.

The calculations were performed with the ENDF/B-VI.2 and VI.8, JEF2.2, JENDL3.2 and JENDL3.3 libraries. Only the main results are summarized here. The details are given in Ref. [3].

The most significant observations are that:

(a) The best overall agreement with the experiments is given by the JENDL3.3;
(b) For homogeneous MOX lattices all but JENDL3.3 libraries give a lower result than for UO_2 lattices;
(c) For cases of MOX fuel with absorber rods, all libraries underestimate the experimental results, the ENDF/B-VI libraries by more than one per cent, the JENDL libraries by about half of one per cent.

A trend analysis was made with several variables using the USLSTAT code for the experiments with MOX fuel in homogeneous lattice [1]. These variables were the uranium enrichment (ratio of the U-235 atoms to total number of uranium atoms), plutonium enrichment (ratio of fissionable plutonium atoms to the total number of plutonium atoms), H/X (ratio of number of hydrogen atoms and fissionable atoms), lattice pitch, outer clad diameter, and the energy of average lethargy causing fission. For these parameters, the correlation coefficients for the calculated k_{eff} values were evaluated, and if the correlation was statistically significant the USL due to this parameter was calculated. Finally, the most restrictive USL was determined. For MOX fuel using the JENDL 3.3 library, this USL is 0.937.

To obtain a qualitative picture of whether these experiments are similar to the application of interest (i.e. the compact storage rack and the TK-6 container), some simple physical quantities related to the multiplication factor were calculated for the experiments and for these applications. These were the fission, capture and neutron fluxes calculated in 5 energy groups with group boundaries 0.1 eV, 10 eV, 10 keV, 100 keV and 20 MeV. The relative importance of different isotopes in fission and capture processes was also investigated. This comparison shows that these broad group characteristics of the applications are covered fairly well by the experiments. Unfortunately, this is not the case for the relative importance of the uranium and plutonium in fission and capture. The ratio of number of fissions on plutonium and number of fissions on uranium, as well as a similar ratio for capture, were evaluated for the benchmarks and for the two burnup credit applications. These ratios are higher by factors of 5–20 for fission and by factors of 3–10 in the benchmark experiments than in the burnup credit applications. This suggests that the uranium has a much lesser role in the selected MOX experiments than in the investigated burnup credit application. Consequently, if a USL is derived by the traditional method from these experiments

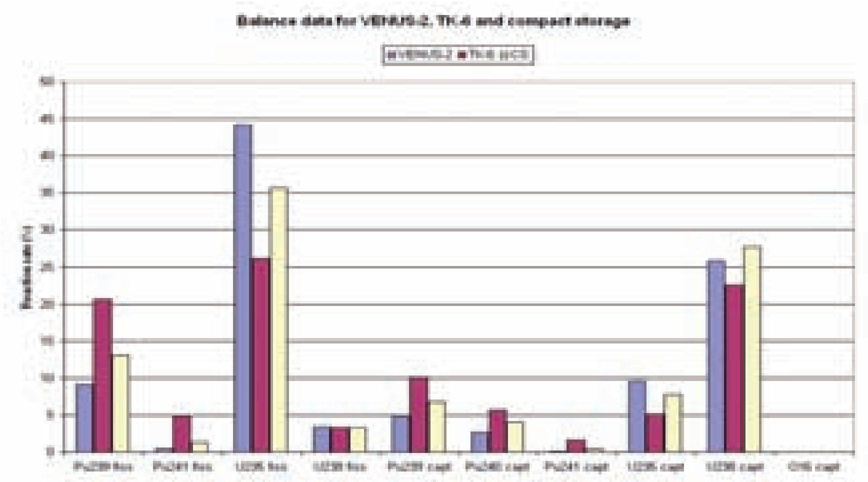

FIG. 3. Distribution of fission and capture on different isotopes in the VENUS-2 MOX core experiment.

only, the uncertainties associated with the plutonium would be overemphasized and the uncertainties associated with the uranium would be underemphasized. The combination of these benchmark sets with other experiments may be useful, but further considerations are needed. Application of sensitivity/uncertainty analysis developed recently at Oak Ridge [4, 5] and implemented in the new version of SCALE [8] may solve this problem.

There are only a few publicly available MOX experiments in which the uranium enrichment corresponds to a burnup credit application. Such an example is the VENUS-2 MOX core experiment. Its details were published as part of an OECD/NEA benchmark study. The geometry of this experiment is quite different from the burnup credit application under consideration because the core consists of three parts, each filled with different fuels: 3.3% enriched UO_2 fuel, 4.0% enriched UO_2 fuel and with MOX fuel containing 2.0% enriched uranium and 2.7% plutonium oxide. The plutonium in the MOX fuel is reactor grade quality so its fissile isotope content corresponds to the desired applications. In spite of the essential difference in the geometrical arrangement it can be shown, that averaging over the whole core, the particular isotopes play a similar role as in the two investigated applications. The balance of fission and capture on different isotopes is shown in Fig. 3, averaged over all fuel pins (fission + capture normalized to 100 for all fuel). It can be seen that the reaction rates for different isotopes are not far from the values found in the applications. It would be of great value for burnup credit users if some other MOX configurations of VENUS were available via the OECD/NEA.

5. TESTING OF THE DEPLETION MODULES OF THE KARATE CODE SYSTEM BY MEASUREMENTS

Experimental validation of depletion calculation methodology is an essential step in the burnup credit licensing process. However, in the case of WWER fuel there is a lack of post-irradiation experimental data. A few journal articles were published in the Russian Federation in the early 1980s, containing experimental data for the WWER-440, but the sets of measured isotopes for burnup credit applications were incomplete and the data appear not to have been well documented. Only recently, supported by the USA and reflecting the urgent need for such data, was a WWER-440 post-irradiation experiment relevant to burnup credit applications performed in Dimitrovgrad. This post-irradiation experiment yielded data for the WWER-440 spent fuel actinides and major fission products in WWER-440 spent fuel and can be used by the entire WWER criticality safety community without restrictions.

In the experiment, an assembly with 3.6% enrichment was selected from the Novovoronezh 4 unit. The assembly was irradiated over four cycles. Eight samples were cut out from four fuel pins. Due to the selection of the pins and the axial positions of the samples, the burnup of the samples ranges from about 28 MW·d/kg U to 49 MW·d/kg U. The concentration of the actinides and major fission products in the samples was measured and published in Ref. [6].

Detailed operational histories and reloading patterns were published for only the four cycles in which the samples were irradiated and the orientation of the assembly during irradiation was not specified. These data are insufficient for calculation of the local irradiation history and the subsequent calculation of composition. The local irradiation histories calculated by the Russian BIPR and PERMAK codes were described in Ref. [6]. The temperature, power density and boron concentration were provided as input data for the KARATE depletion module. KARATE is a coupled neutron physical–thermohydraulical code developed at KFKI AEKI [7]. The concentrations of U-235, U-236, U-238, Pu-238-242, Am-241, Am-243, Nd-143, Nd-145, Cs-133, Sm-147, Sm-150, Sm-151, Sm-152, Mo-95, Tc-99, Ru-101, Ag-109, Gd-155 were calculated. Comparison with measurements is given as the ratio of the calculated and measured values in Figs 4–7.

As can be seen from the figures, the agreement between the calculated and measured values is not very good. There are very significant deviations for important isotopes. Large deviations between calculation and measurements were found in the case of samples from the pin at the corner of the assembly. This may be connected to the error in the calculation of close heterogeneity. It is important to note that some participants of the workshop on the Need for Post Irradiation Experiments to Validate Fuel Depletion Calculation

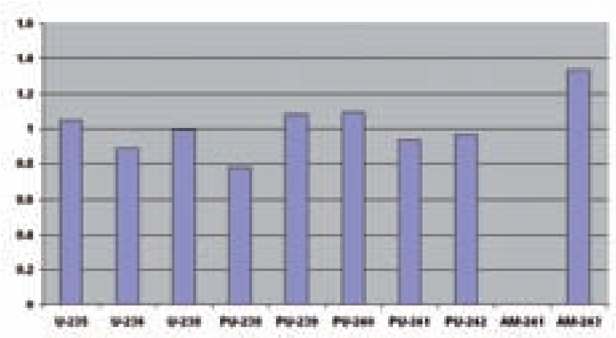

FIG. 4. Average values of calculated/measured ratios for actinides.

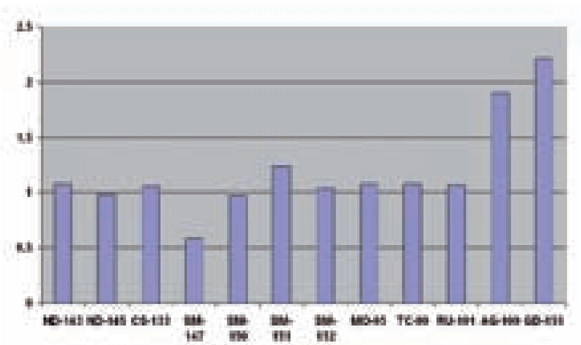

FIG. 5. Average values of calculated/measured ratios for fission products.

Methodologies, held at the Nuclear Research Institute Rez, Czech Republic, from 11–12 May 2006, found similar discrepancies. The explanation of these discrepancies should be investigated.

6. SUMMARY

The impact of new fuel types on the subcriticality of the storage/transport facility used in Hungary has been summarized. It was found that for the compact storage pool and for the TK-6 transport cask, the subcriticality criteria can be met only with capacity reduction if the fresh fuel assumption is used. The degree of burnup required for meeting the subcriticality criteria was determined. The possibility of using the available experimental data for

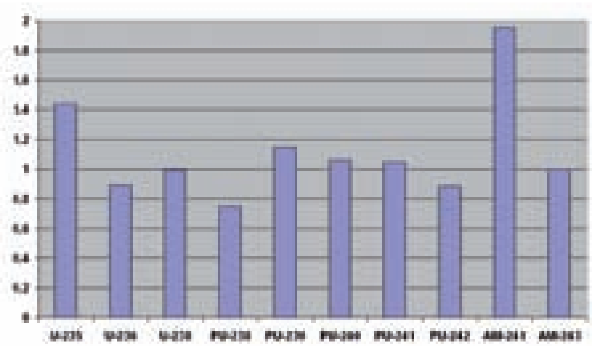

FIG. 6. Values of calculated/measured ratios for actinides in the case of a sample with maximum burnup.

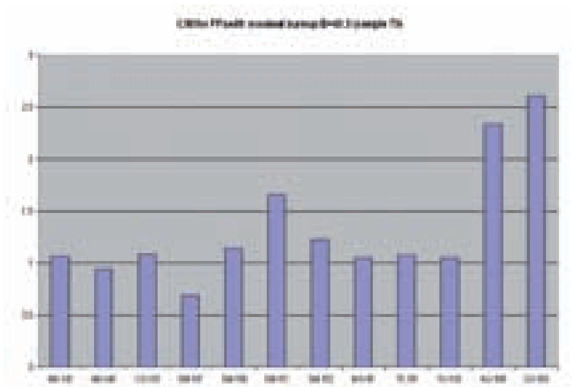

FIG. 7. Values of calculated/measured ratios for fission products in the case of sample with maximum burnup.

determining the uncertainty of criticality and depletion calculation was investigated.

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THE SOUTH AFRICAN EXPERIENCE IN SPENT FUEL MANAGEMENT AT THE KOEBERG NUCLEAR POWER PLANT — THE PERSPECTIVE OF THE NATIONAL NUCLEAR REGULATOR

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Abstract

The paper summarizes the experience of the National Nuclear Regulator in the licensing of the reracking project at the Koeburg nuclear power plant in South Africa. The licensing took approximately five years from the initial project meetings between the operator (Eskom) and the regulator. The implementation of the project was delayed considerably due to various factors, such as safety requirements influencing the design of the racks and plant requirements, and problems with the manufacturing process as well as the packaging and transport of the racks. Due to the delays in the spent fuel pool reracking project, Eskom had to develop a contingency plan, which involved the use of the dry storage casks. The paper describes the various stages in the licensing process, including the steps taken to ensure that subcriticality is maintained in the new system.

1. INTRODUCTION

In 1990, in anticipation that transport from the Koeberg nuclear power plant spent fuel pools to a remote interim dry storage site would take place, Eskom (the South African Electricity Utility and Operator of the Koeberg plant) placed an order for four Castor type X/28 F casks licensed for dual transport/storage. The casks also had the capability of functioning as a contingency store with the capacity for 112 spent fuel assemblies. In parallel, Eskom applied to the National Nuclear Regulator (NNR) for permission to commence reracking the spent fuel pools so as to accommodate all the spent fuel that would accrue during the remaining lifetime of the plant. This was considered to be possible due to advances in criticality depletion codes and by taking credit for spent fuel burnup and the soluble boron present in the spent fuel pools. These considerations would make it possible for the storage capacity of the existing storage pools at Koeberg to be almost doubled. Due to delays in the spent fuel pool reracking project, Eskom had to develop a contingency plan which involved the use of the dry storage casks.

2. OVERVIEW OF THE RERACKING PROJECT

The plan to accommodate the storage requirements at Koeberg for a further period of 40 years involved the use of super high density racks involving the use of borated stainless steel as a neutron absorber to ensure subcriticality. The added advantage of this option is that the borated stainless steel does not shrink or swell (a problem encountered with the existing racks) and exhibits better corrosion resistance than the existing racks. The plan included taking credit for fuel burnup, which resulted in a two storage region pool, namely region I and region II.

Region I consists of three high density rack modules. There are no restrictions relating to the burnup of fuel assemblies and the boron content of pool water. Within this region, 210 cells are provided to store unirridiated fuel with a maximum initial enrichment of 5% U-235. This means that it can accommodate one complete core load (157 fuel elements) plus one fresh fuel batch (53 fuel elements).

Region II consists of high density storage racks designed to receive spent fuel assemblies that have undergone the minimum burnup to meet the acceptance criterion for storage in this region. It consists of 14 rack modules with a total of 1326 spent fuel storage locations.

3. LICENSING STAGES

Project licensing was conducted in accordance with the NNR internal procedures STI-20, Review Plan for Safety Assessments Associated with Koeberg Nuclear Power Station. In accordance with this procedure, the following steps are required:

(a) Initial notification to NNR;
(b) Description of the project;
(c) Scope of the safety case;
(d) Installation phase of the project in the spent fuel building;
(e) Decommissioning of old racks;
(f) Operational phase.

It was planned that the new storage racks would be installed prior to refuelling outages 112 and 212. However, due to a lack of sufficient storage space it was proposed to use two of the Castor X/28F dry storage casks per unit to ensure that there would be sufficient free fuel storage locations to shuffle fuel in order to remove the existing racks and install the new racks. The licensing of the Castor X/28F casks is described in Section 5.

3.1. Licensing of high density racks

In order to provide a framework within which to license the spent fuel reracking project, a three phase licensing strategy was adopted:

— Phase 1: Installation and use of the racks with restriction on the number of fuel assemblies to 772 and 733 (unit 1 and unit 2 pools, respectively) until additional structural support is provided to the spent fuel pools. This is to account for the additional weight of fuel assemblies because of the increased capacity and the additional weight of the new racks. This includes a partial upgrade of the spent fuel crane (PMC) with the necessary interlocks to prevent fuel misplacement and of the spent fuel pool cooling system (PTR) configuration to remove additional heat.
— Phase 2: Construction of additional support structure for the pools to allow operation at full capacity of the pools (1536 fuel elements per pool). Full upgrade of the PMC crane with software to control fuel movement. Further upgrade of the PTR system to comply with ANSI 57.2 requirements [1].
— Phase 3: Upgrade of the PTR system with increased redundancy and optimized compliance with ANSI 57.2 requirements [1].

3.2. Technical aspects

The licensing strategy addressed the adaptation of ANSI 57.2 [1] standard design requirements. With ANSI 57.2, credit can be taken for spent fuel burnup to ensure a $K_{eff} < 0.95$. Under certain conditions, a criticality accident may therefore be possible in the spent fuel pool. This was not the case under the original design code ANSI N 18.2, to which the Koeberg spent fuel pools were built.

Compliance with safety functions as specified in the design code had to be demonstrated in the project safety case during all phases of the project (during installation and the operational phase). The following main safety functions were defined:

(a) Ensure that the fuel assemblies maintain their margin to criticality;
(b) Ensure adequate cooling of the stored fuel;
(c) Ensure that the risk of fuel damage is acceptable.

Adaptation of the ANSI 57.2 standard had further implications for the spent fuel storage capacity. Many boundary modifications had to be implemented, which are described further in this paper.

3.2.1. Structural design

Each fuel pool is equipped with three region I and 14 region II high density storage racks. The storage cells are designed to accommodate 17 × 17 type fuel assemblies. The racks are designed to allow sufficient coolant flow to remove the decay heat generated by the fuel assemblies and to ensure the required subcritical configuration under all specified plant conditions. All racks are classified as seismic category I components, and are therefore required to remain functional during and after a safe shutdown earthquake.

Region I rack modules are formed by assembling structural cells of unborated stainless steel channels in an edge to edge configuration so that they enclose the fuel assemblies along their entire length in a checkerboard array. Borated stainless steel poison boxes (1.7–2% natural boron) are inserted as the neutron absorber in the cells. The boron content and thickness of the boxes are such that fresh fuel, up to an enrichment of 5% U-235, can be stored in the racks. The complete structure is bolted on to a baseplate. The free standing rack modules rest on adjustable pads on top of the load distribution plates. It is ensured that sufficient cooling water flows into and through the racks by openings through the perforated bottom baseplate.

Region II racks are built to the same principles as the square stainless steel channels, but the borated stainless steel poison sheets are attached inside the cells so that the fuel assemblies are always separated by a borated steel sheet. The remainder of the construction is the same as described for region I.

The racks are set down without anchoring on the bottom of the pool. They are separated from each other by small water gaps. Fluid–structure interaction phenomena reduce the dynamic forces in the event that the racks collide during a seismic event. The seismic analysis (dynamic and stress analysis) was performed by the rack designer and the review of structural and seismic design of the spent fuel storage racks was independently performed by consultants. Several additional analyses were performed on different aspects of the conceptual design in order to verify the new design conditions that were introduced by the design and/or manufacturing changes or new design parameters.

3.2.2. Criticality evaluation

In support of the ultra high density spent fuel racks designed for Koeberg, the designer proposed a methodology by which the criticality safety of the reracked pool could be demonstrated. The fundamental premise of the methodology is that credit may be taken for the burnup of the assemblies to be stored, thereby introducing many additional calculational considerations in comparison with a 'fresh fuel' approach. As safety margins are diminished, the accuracy of data used and the remaining conservatisms must be well defined. A factor that emerged after the assessment of the methodology was that the existing methodology was inadequate in that credit for burnup alone was not sufficient to demonstrate the subcriticality of the pool under the standard acceptance criteria. Introduction of additional anti-reactivity into the pool was thus required and the licensee, Eskom, opted for taking further credit for a minimum concentration of soluble boron rather than inserting solid neutron poisons into assembly guide tubes.

The design of region I of the storage pool is based on the assumption that the fuel loaded into the racks is fresh and unpoisoned and with a maximum enrichment of 5%. The design of region II racks is based on the assumption of a minimum fuel burnup on the initial enrichment of the fuel stored. The decision criterion is based on the reactivity equivalence curve — any spent fuel element that does not satisfy this condition has to be stored in region I or in a checkerboard arrangement in region II. The criticality safety analysis was performed by the designer and satisfied the requirements of the ANSI 57.2 standard [1]. As required by this standard, the evaluated multiplication factor (k_{eff}) of a storage configuration shall include all uncertainties arising from the applied calculation procedure and from manufacturing tolerances and shall maintain a margin to criticality of 0.05 k_{eff}. All uncertainties are expressed as a 95% probability and a 95% confidence tolerance limit. The criticality analysis, proving that sufficient margin to criticality is maintained, is carried out for the most reactive fuel assembly type and the most reactive storage condition. The determination of the most reactive storage condition included, among others, evaluation of abnormal and accident conditions, e.g. displacement or deformation of the channels caused by seismic events, drop of fuel assembly or heavy object during handling, inadvertent placement of fresh fuel assembly in region II of the storage rack, and so on. For plant condition IV events (in accordance with ANSI 57.2), credit was taken for a minimum boron content in the spent fuel storage pool.

The NNR conducted an in-depth assessment of the analysis submitted to demonstrate subcriticality of the spent fuel pool during all phases of operation and for credible accident situations in accordance with ANSI 57.2. The main initial concern of the NNR was the credit given in the analysis for neutron absorbing fission products in the fuel assemblies. After many deliberations, the licensee (Eskom) decided to adopt the approach of using partial credit for boron and taking credit for burnup. This approach necessitated analysis of a boron dilution accident to support the design basis.

The most important criticality issue, which resulted from the use of burnup and partial boron credit, was that of localized criticality. Various neutronic effects had to be considered, such as the end effect resulting from the axial burnup shape, the operating temperature profiles under which this axial shape was obtained and the assumptions made in order to model these important effects. It was shown by the NNR that a local dilution of less than two cubic metres of unborated water could invalidate the assumptions made in the safety case. The question was whether a slug of unborated water could enter through the PTR return line and remain in a very slightly diluted state for long enough to reach the top part of the fuel assemblies, in which case the safety case for the high density racks would have been jeopardized. After alternative calculations performed for the NNR, the conclusion was reached that because of the low flow rate of water through the fuel assemblies due to convection and the relatively high flow rate of the spent fuel pond cooling system return line, it would be incredible to have a slug of water with a boron concentration less than 440 ppm on top of the centre fuel assemblies. Additionally, by means of a probabilistic safety assessment study, Eskom intended to show that the local dilution was incredible. The NNR, however, considered that there were too many questionable assumptions, mainly related to human factors.

Based upon all of this work the NNR decided to consider the scenario of localized criticality in the Koeberg spent fuel pool as credible. A number of modifications were identified to address the issue, including installation of spargers to spray make-up water into the pool.

3.2.3. Spent fuel pool cooling

In order for the Koeberg plant to comply with the additional design requirenents as specified by the ANSI 57.2 standard [1], it was agreed that a phased upgrade of the spent fuel pool cooling system would be applied. In the first phase, an upgrade of the instrumentation to provide redundant operator indications and a dedicated water make-up system were required. In the final phase, an independent spent fuel cooling system with higher heat removal capacity will be installed (currently in the final design stage).

The bounding case thermohydraulic analysis was performed by the designer and independently reviewed by an independent contractor. The input assumptions included a full core unload five days after reactor shutdown, with all fuel storage locations filled by previously discharged fuel with fuel enrichment assumed to be 5.0%. The analysis demonstrated that the spent fuel pool temperature would remain below the limit of 50ºC as specified in the Koeberg Operational Technical Specifications, and that no localized boiling would occur.

4. RACK INSTALLATIONS

The safety requirements stated in ANSI 57.2 [1] and NUREG 0612 [3] were considered during the installation phase of this project. Precautions to prevent fuel damage were included in the design of the 20 tonne temporary installation crane, in the design of the handling and lifting devices, as well as in the development of detailed work plans and procedures. The layout of the 20 m level of the spent fuel building was designed to enhance both nuclear and personnel safety at all times during installation. The installation safety case was prepared, in which the following main key elements were addressed to ensure nuclear safety:

(a) Safe load paths in all handling operations;
(b) All necessary shielding provided for all operations during rack installation;
(c) Criticality control in shuffling fuel from the old to the new racks;
(d) Provisions made (and special tools provided) for rack ‘settling’ during first fuel loading;
(e) Capability for emergency core unloading to be maintained at all times;
(f) Provisions for transferring a dry storage cask to the cask loading cell if the need arises;
(g) No fuel movement from region I to region II allowed until completion of PMC crane upgrade;
(h) Analysis of all possible incidents (load drop) considered, consequences and recovery actions analysed.

5. LICENSING OF CASTOR CASKS

Due to delays in the spent fuel pool reracking project, Eskom had to develop a contingency plan which involved the use of the four Castor type X/28F dry storage casks. These casks would provide enough storage space to delay the need for high density storage racks by one fuel cycle on each unit and also ensure that there are sufficient free storage locations to shuffle fuel in order to remove the existing racks and install the new racks.

The decision was taken that the licensing of the dry storage casks would be performed in two phases:

— Phase I — approval in principle for use of casks based on the demonstration that the casks meet the applicable safety criteria;
— Phase II — approval of cask loading and transport to, and storage in the storage building, based on the demonstration that the appropriate nuclear safety criteria were met.

The licensee was required to produce a safety case for each of the two licensing stages.

5.1. Cask licensing — Phase I

The NNR review of the safety case of Phase I concentrated on the following design principles:

— Structural — the casks are designed, fabricated and tested to maintain the confinement of the fuel assemblies, both for normal operation and under accident conditions, as defined by the IAEA [2];
— Containment — the casks have to be leaktight during storage and transport to prevent any gas leakage from the cask;
— Shielding — maximum allowable total dose at any point on the cask surface and at a distance of two metres is within the limits specified in Ref. [2];
— Criticality control — the effective multiplication factor (k_{eff}) is below 0.95, including uncertainties;
— Fuel integrity — peak cladding temperature of the hottest fuel rod (hottest part of the fuel rod) does not exceed 340ºC at 38ºC ambient air at outer cask environment under storage conditions.

After a detailed review, the NNR granted approval for phase I of the project and issued a certificate under the condition that loading of the fuel would only be allowed once phase II of the project had been approved.

5.2. Cask licensing — Phase II

The final safety case of phase II was submitted to the NNR and the following issues were evaluated:

(a) Loading of spent fuel from the spent fuel pool into the casks and associated cask handling operations, with the specification for any assembly to be loaded into a cask to include 10 years storage in the spent fuel pool and enrichment not exceeding 3.25%;
(b) Identification and qualification of a building to store the casks;
(c) Transport of the loaded casks to the storage building;
(d) Monitoring and maintenance of the casks within the storage building;
(e) Security and access control of the casks;
(f) Subsequent transport of the loaded casks from the cask storage building to the spent fuel pool, fuel unloading and associated cask handling operations.

The NNR assessment of these aspects concluded that all licensing requirements were met and therefore the safety case was accepted.

6. BOUNDARY MODIFICATIONS

In order to install the new racks, many modifications to the plant had to be implemented, such as installation of a 20 tonne temporary crane, installation of a suction strainer on the spent fuel pool pond cooling system (PTR) suction line, cutting of the PTR discharge line, installation of the liner support plates, and modifications to the PMC crane.

7. CONCLUSIONS

The licensing aspects of the reracking project took approximately five years from the initial project meetings between the NNR and Eskom to the approval of the final safety case. The implementation of the project was delayed considerably due to various factors such as safety requirements influencing the design of the racks and plant requirements, and problems with the manufacturing process, as well as the packaging and transport of the racks. In spite of the above problems, the implementation phase of the reracking project was conducted relatively smoothly.

The installation of the racks was essential to the continued operation of the Koeberg nuclear power plant and has been demonstrated to be safe. The NNR acceptance of the final safety case for the project was based on the following:

(a) The safety rationale for the design, installation and subsequent use of the high density storage racks were adequately addressed in the safety case after intervention by the NNR;
(b) The total risk to the operators and the public during the installation and operational use of the racks was quantified and found to be acceptable.

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NEW APPROACH TO NUCLEAR SAFETY ASSESSMENT OF THE WWER-440 REACTOR POOL*

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Abstract

A criticality analysis of the reactor pool for existing fuel types for the WWER-440 reactors has been carried out according to the requirements of existing regulatory documents in force in Ukraine. It was concluded that some of the regulatory requirements are too conservative in view of current international practice. The paper points out the areas of conservatism and makes recommendations for taking account of the burnup credit system used in several other countries.

1. INTRODUCTION

Preliminary analysis of the regulations in force in Ukraine concerning the safety of spent nuclear fuel management systems shows that some regulatory requirements in force are too conservative when compared with current international practice.

The extent of the conservatism can be determined and reduced, if necessary, using calculations to analyze the criticality status of spent nuclear fuel management systems. Such an activity is consistent with the state of the art developments in the field. However, it must be based on improved understanding of the processes occurring in nuclear storage systems and on improved capabilities with regard to accuracy, correctness and reliability in the numerical modelling of these processes.

This work is intended to demonstrate that the excessive conservatism used previously in relation to the requirements for criticality safety in Ukraine can be considerably decreased through the use of more realistic modelling of fuel systems. If such modelling is performed with the use of state of the art computer codes, based on a more complete understanding of the processes in

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* Work carried out with the financial support of the US Government.

fuel systems, then removal of the excessive conservatism will not reduce the safety of nuclear fuel storage systems.

Criticality safety has been analysed according to the regulatory requirements in force, taking into account the conditions under which a system for storage and management of nuclear fuel has the maximum k_{eff} value pursuant to the following requirements:

(a) Consideration should be given to the errors associated with the methods of calculation, concentrations, isotopic content of absorbers and manufacturing tolerances;
(b) The presence of a reflector should be assumed.

Most of the criticality calculations presented in this paper have been obtained with the SCALE code package [1]. Some verification and confirmation calculations have been done with the MCNP program [2].

The isotopic content of the spent nuclear fuel was calculated using the German cell code NESSEL [3]. This code has been used for the past several years to analyse the fuel loading of Ukrainian nuclear power plants for the determination of neutron-physical constants for WWER fuel, depending on burnup. Both SCALE and NESSEL have previously been subjected to comprehensive testing for acceptability to WWER fuel calculations [3–5].

2. INPUT INFORMATION FOR CALCULATIONAL STUDIES

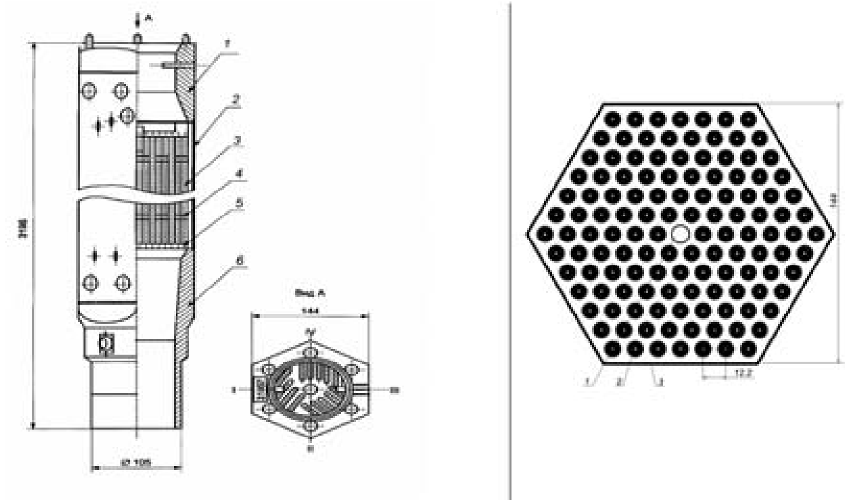
2.1. Fuel elements of the WWER-440 reactor

A working fuel assembly (Fig. 1) consists of a bundle of fuel pins, head, tailpiece and hexagonal cover. Fuel pins in the bundle are set into a triangular grid with a (12.2 ± 0.15) mm pitch. The basic technical characteristics of the working fuel assembly are shown in Table 1 [6].

2.2. Reactor pool of WWER-440 units

The reactor pool is located in the steam generator box and is connected at its upper part to the reactor cavity and refuelling vault by refuelling channels. Both channels are equipped with hydro-seals to isolate the pool water from the reactor cover and refuelling vault [7].

The reactor pool stores both sealed and damaged spent fuel assemblies. Sealed spent fuel assemblies are stored directly in reactor pool rack cells and damaged ones are stored in hermetic canisters installed in specially designed



1: FA head; 2: cover; 3: fuel pin; 4: spacer grid; 5-lower grid; 6: tailpiece

1: central tube; 2: fuel rod (fuel pin); 3: assembly cover

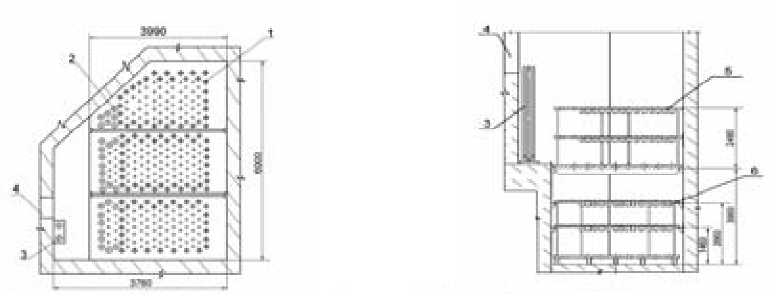
FIG. 1. Scheme of the grid for fuel rods of the WWER-440 fuel assembly.

cells. Fuel must be stored in the reactor pool under a water layer with a boric acid concentration of not less than 16 g/kg (approximately 2800 ppm). The water level in the reactor pool must be not less than 7.3 m and not less than 3000 mm above the assembly heads. The water in pool compartments circulates through the cooling system, keeping the water temperature during storage to not more than 50°C and not more than 70°C in the case of full unloading of the core fuel. The pool walls are made of concrete. To avoid leakage, the pool has a double layer. The coating material is corrosion resistant acid proof steel. The reactor pool racks (main and removable) (Table 2) are designed for the storage and cooling of leak-tight spent fuel assemblies and hermetic canisters containing the assemblies which have shown some damage/leakage. The rack sections are metal structures that consist of a lower support plate and two spacer grids connected by poles (Fig. 2).

The lower (main) rack of the reactor pool contains 319 cells for fuel assemblies and 60 cells for hermetic canisters. 296 cells for assemblies and 54 cells for hermetic canisters are placed on the upper (removable) rack. Thus a total of 729 spent fuel assemblies can be placed in the reactor pool. The upper rack is divided into three sections. It is assembled as the second stage over the main rack.

TABLE 1. WORKING ASSEMBLY PERFORMANCES

Core element	Value
Overall length of working fuel assembly (mm)	3217
Length of assembly fuel part (height of fuel column) (mm)	2420
Wrench dimension of assembly in assembly head area (mm)	144
Enrichment of UO_2 fuel (%)	3.6, 4.0, 4.4 (±0.05)
Maximum burnup (GW d/t U)	53.5
Mass of fuel (uranium dioxide) (kg)	139.96 ± 2.77
Wall thickness of assembly cover (mm)	1.5
Number of spacer grids	11
Central tube, outer diameter × thickness (mm)	10.3 × 0.75
Number of fuel pins in assembly	126
Grid pitch of fuel pin (mm)	12.2
Outer diameter in fuel pin (mm)	9.1
Materials of assembly elements:	
Fuel pin cladding, central tube and spacer grids	(Zr + 1% Nb)
Assembly cover	(Zr + 2.5% Nb)
The rest of the assembly elements	'08X18H10T' steel



(a) Schematic view of RP horizontal cross-section (b) Schematic view of RP vertical cross-section
1: cells for fuel assemblies; 2: cells for hermetic canisters; 3: SDDA canister; 4: refueling channel; 5: removable rack; 6: main rack

FIG. 2. WWER-440 reactor pool racks.

The assemblies and hermetic canisters are placed in racks in the form of a triangular mesh with a pitch of 255 mm. For effective removal of residual assembly heat there are apertures in the lower plate which provide for cooling water circulation both inside and between the assemblies. The parts and units of the reactor pool racks are made of corrosion resistant acid proof steel.

TABLE 2. MAIN CHARACTERISTICS OF THE WWER-440 REACTOR POOL

Parameter	Value
Fuel storage racks: length × width × height (mm)	6000 × 3990 × 2480
Distance from pool bottom to upper plate of lower rack (mm)	2660
Distance from pool bottom to lower plate of upper (removable) rack (mm)	3980
Total number of cells, pieces	729
Number of cells for working assemblies: in upper (removable)/lower rack, pieces	296/319
Number of cells for hermetic canisters: in upper (removable)/lower rack, pieces	54/60
Fuel assembly pitch (mm)	225
Material	steel

3. CALCULATION RESULTS FOR THE REACTOR POOL OF THE WWER-440 UNIT

This section presents results of the criticality calculation for the reactor pool of the WWER-440. The main criterion for criticality safety in such calculations is the requirement of Ref. [1] that reactor pool subcriticality assurance should be 5% ($k_{eff} < 0.95$) for the states with the maximum neutron multiplication factor k_{eff}.

The factors affecting k_{eff} have been considered and analysed in the selection of the calculational model. The calculations have been carried out as follows:

(a) Calculation of reactor pool compartment criticality due to water density changes in it. Numerical comparative calculations were conducted to determine the correctness of the results obtained.
(b) Calculation of reactor pool criticality giving credit for fuel burnup and/or changes in the boron acid concentration in the pool cooling water. The water density was also varied from 0 to 1 g/cm^3.
(c) Calculation of reactor pool criticality for partial loading.

The calculations performed and the results obtained are described below in more detail.

3.1. Description of calculation parameters and results obtained

Two types of fuel are used in the WWER-440 reactor — the fuel assembly (maximum enrichment is 3.6% of ^{235}U) and the working assembly (maximum enrichment is 4.4% of ^{235}U).

From a criticality analysis perspective, a fuel assembly differs from a working assembly with the same fuel enrichment only because of its lesser height, and therefore it contains less fuel. Thus, conservatively, the criticality analysis is performed for working assemblies only. The highest level of fuel enrichment (4.4%) was selected for the criticality analysis, taking into account an enrichment inaccuracy of 0.05%, which can appear during manufacturing (in other words 4.45% enrichment has been assumed). Also, it has been assumed that there is an inaccuracy of ±2.77 kg per assembly in fuel mass due to the calculation of the effective fuel density (which consists of complex uranium pellets). For other assembly characteristics the design values were applied since their tolerances did not cause significant changes in k_{eff}. All calculations have been performed for a fixed number of histories (1000 generations of 1000 neutrons — in other words, 10^6 histories); this gives a statistical inaccuracy of $\sigma \approx 0.001$ in k_{eff}.

In such conditions, under the design parameters of the reactor pool the criticality of the reactor pool, for normal operating conditions (that is, the boron acid concentration in the pool water is 16 g/kg) is:

$$k_{eff} \pm \sigma = 0.5161 \pm 0.0006$$

In accordance with regulatory requirements [1] for storage facilities with a homogeneous absorber it is necessary to assume, while analysing safety in spent nuclear fuel storage, that there is no absorber. Therefore, in all subsequent calculations, if not otherwise stated, the presence of boron acid in reactor pool cooling water is not credited.

Table 3 presents the calculation results for k_{eff} of the reactor pool loaded with an assembly of 4.4% enrichment, depending on the cooling water density. k_{eff} rapidly increases in conditions of decreasing water density in the pool. As Table 3 shows, at a water density of 0.8 g/cm^3 the requirements of Ref. [1] are not met and the system becomes overcritical. The peak of the k_{eff} value is for the air–water mixture density at 0.2–0.25 g/cm^3. Since the density at 0.25 g/cm^3 leads to a higher result than in subsequent analysis, it is applied.

Part of the calculations were replicated using the MCNP programme [2] in order to increase the reliability of the results obtained and for comparison with independent results. These data are also given in Table 4 and demonstrate the high consistency level obtained in comparison with SCALE results. The

TABLE 3. DEPENDENCE OF k_{eff} ON WATER DENSITY IN THE WWER-440 POOL

g/cm^3	0.0	0.1	0.2	0.25	0.3	0.4	0.5	0.6	0.8	1.0
k_{eff}	0.6515 ± 0.0005	1.2222 ± 0.0008	1.3376 ± 0.0007	1.3394 ± 0.0007	1.3258± 0.0007	1.2667 ± 0.0007	1.1970 ± 0.0007	1.1241 ± 0.0007	1.0038 ± 0.0007	0.9277 ± 0.0008

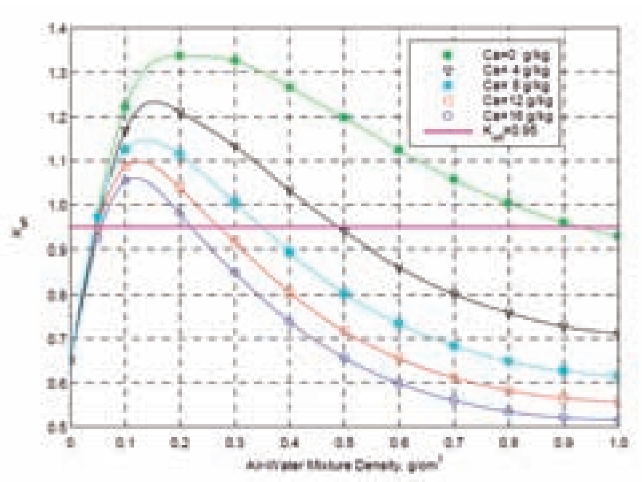


FIG. 3. k_{eff} dependence on air–water mixture density in different boron acid concentrations in WWER-440 RP.

results show that the SCALE calculations in most cases provide for more conservative values of the neutron multiplication factor (within 1%).

Figure 3 shows the dependence of the neutron multiplication factor on the air–water mixture density for different boron concentrations in the reactor pool. As shown in this figure, the system is supercritical for the whole range of possible concentrations of boron acid — from shutdown (2800 ppm) to unborated water at low densities of the moderator. Also, there is a shift to the maximum from 0.2–0.25 g/cm^3 for unborated water to 0.1 g/cm^3 for the shutdown boron acid concentration.

From Fig. 3 the maximum value of k_{eff} and appropriate air–water mixture density for each value of the boron concentration in the reactor pool can be determined (Table 5).

TABLE 4. k_{eff} DEPENDENCE ON WATER DENSITY IN WWER-440 RP (WITH NO BORIC ACID IN THE REACTOR POOL WATER)

No.	Enrichment ^{235}U (%)	Mass of ^{235}U in assembly (kg)	Temperature, (K)	Density (g/cm^3)	k_{eff}		
					OKB Hydropress (Moscow, Russian Federation)	SSTC NRS (Kiev, Ukraine)	
						SCALE	MCNP
1	3.65	122.7	400	0.958	0.9424 ± 0.0011	0.9521 ± 0.0007	—
2	3.6	120.2	300	0.958	0.8987 ± 0.0031	0.9075 ± 0.0007	0.8985 ± 0.0007
3	4.21	120.2	300	0.998	0.9160 ± 0.0054	0.9176 ± 0.00087	0.91725 ± 0.0007
4	4.21	120.2	300	0.958	0.9202 ± 0.00098	0.9293 ± 0.0008	0.9298 ± 0.0007
5	4.21	120.2	300	0.7	1.0450 ± 0.0058	1.0508 ± 0.0007	1.0510 ± 0.0006
6	4.21	120.2	300	0.25	1.3475 ± 0.0055	1.3455 ± 0.0007	—
Boundary conditions			Upper part: 70 cm water; escape. Lower part: 70 cm water, mirror reflection				

TABLE 5. OPTIMUM AIR–WATER MIXTURE DENSITY FOR DIFFERENT BORON ACID CONCENTRATIONS IN THE WWER-440 REACTOR POOL

C_B, g/kg	0	4 (700 ppm)	8 (1400 ppm)	12 (2100 ppm)	16 (2800 ppm)
ρ, g/cm^3 (k_{eff})	0.210 (1.339)	0.154 (1.233)	0.141 (1.147)	0.125 (1.100)	0.117 (1.061)

3.2. Criticality calculations of the WWER-440 pool with fuel burnup credit

Since the results given in Fig. 3 demonstrate that the required subcriticality of the system cannot be achieved, even when credit is given for the presence of boron in the reactor pool water, consideration is given to the potential use of burnup credit. The fuel burnup changes the concentration of fuel isotopes, actinides and some fission products: U-235, U-236, U-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241 and Sm-149, Sm-151.

Figure 4 shows the results of the calculations of the effect of fuel burnup on the criticality of the WWER-440 pool system for fuel of 4.4% initial enrichment in the optimum moderation condition (unborated water density 0.25 g/cm^3). The fuel burnup credit leads to a decrease in the neutron multiplication factor from 1.3394 ± 0.0007 (fresh fuel) to 1.1305 ± 0.0006 (burnup: 50 GW·d/t U). The calculations were also performed for the reactor pool loaded with working fuel assemblies of 4.0%, 3.6% and 2.4% initial enrichment and for burnup of 0 and 50 GW·d/t U. These results are also shown in Fig. 4.

The calculation results demonstrate that for loading with 4.4% initial enrichment, the condition $k_{eff} < 0.95$ is not met even for the case when assembly burnup is equal to 50 GW·d/t U. It is therefore impossible to plot a curve of reactor pool loading under these conditions since, for all possible fuel enrichments which were analyzed (2.4, 3.6, 4.0, 4.4%), at an air–water density of 0.25 g/cm^3 the system is supercritical. An exception is for 2.4% enrichment, but the subcriticality is less than the 5% required by regulatory documents and necessary for plotting a loading curve. Based on this it is possible to conclude that use of fuel burnup credit does not lead to a significant improvement in the situation.

The next step is to consider the cases from Table 5 and to combine both approaches and consider the boron contents in reactor pool water and fuel burnup at the same time (Table 6). In the review of the effect of burnup, the pool loaded only with working fuel assemblies of 4.4% enrichment was considered. This shows that it is possible to reach an acceptable level of subcriticality only by taking simultaneous credit for these two factors — the

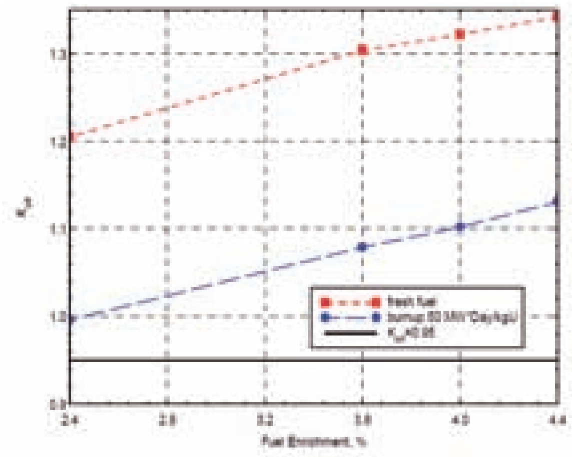


FIG. 4. k_{eff} dependence on fuel enrichment in WWER-440 RP in air–water mixture density at 0.25 g/ccm.

change in the isotope composition and the boron acid presence in reactor pool water.

3.3. Calculations of a partially loaded reactor pool compartment

It is clear from the previous discussion that the requirement of $k_{eff} < 0.95$ is not met for a wide range of possible states of the reactor pool of the WWER-440 unit. If the presence of boric acid in the pool cooling water is taken into account, this decreases the criticality level, but nevertheless does not allow a safe level to be reached.

Calculations have to be performed for emergency unloading of fresh or low burnup fuel from the core. For this purpose, at least the individual reactor pool zones intended for this work should be analysed without fuel burnup credit. Therefore, the next step in the analysis was the review of partial pool loading with fuel assemblies when some of the places are empty or filled with an absorber — absorber rods or spent fuel assemblies (initial enrichment at 4.4%, burnup at 50 GW·d/t U). The calculations were not intended to determine the optimum variant of reactor pool loading, but only to demonstrate the possibility of reducing criticality to the required value.

The calculations considered two variants of fuel assembly layout, row and checkerboard (Fig. 5). These variants were analysed for the pool filled with pure water (air–water mixture) and in the presence of boron acid at a

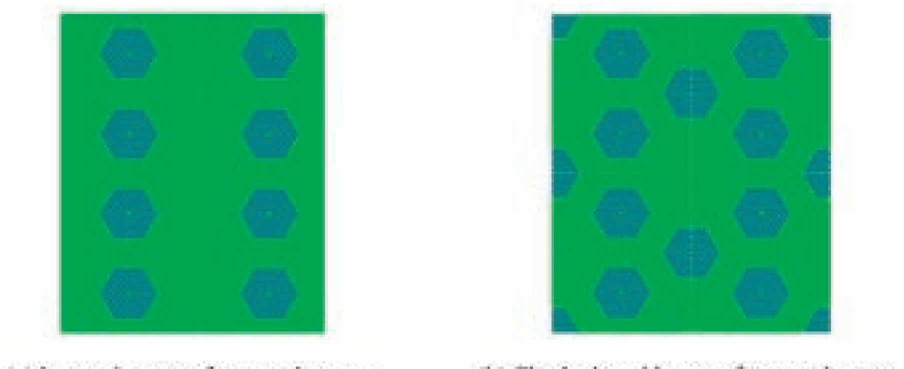


FIG. 5. WWER-440 reactor pool loading.

TABLE 6. MULTIPLICATION FACTOR FOR DIFFERENT BORON ACID CONCENTRATIONS (BURNUP AT 50 GW·d/t U, OPTIMUM MODERATION)

C_B, g/kg	0	4 (700 ppm)	8 (1400 ppm)	12 (2100 ppm)	16 (2800 ppm)
k_{eff}	1.1301 ± 0.0006	1.0092 ± 0.0006	0.9433 ± 0.0005	0.8960 ± 0.0007	0.8590 ± 0.0006

concentration of 4 g/kg (approximately 700 ppm) with a varying air–water mixture density.

The calculation results are shown in Fig. 6. It is seen (Fig. 6(a)) that the requirement $k_{eff} < 0.95$ is not met for either of the variations of partial pool loading with fresh fuel assemblies of 4.4% enrichment. In the case where the pool is filled with borated water, the requirement $k_{eff} < 0.95$ is met for the layout in rows for almost the whole range of air–water mixture densities. For the variant where absorber rods are installed (Fig. 6(b)), the system becomes supercritical only in the range of moderator densities at 0.5–0.1 g/cm^3 and only for the checkerboard layout of the absorber rods in unborated water. The layout in rows of the absorber provides the required level of subcriticality over the whole range of air–water mixture density changes. In the case where burnt fuel is installed instead of the absorber (Figure 6(c)), supercriticality occurs over a wide range of moderator densities. Boron acid injection does not lead to a significant improvement in the results.

The calculations were performed for the complete range of the air–water mixture density. Hence the complete pattern of processes in the reactor pool

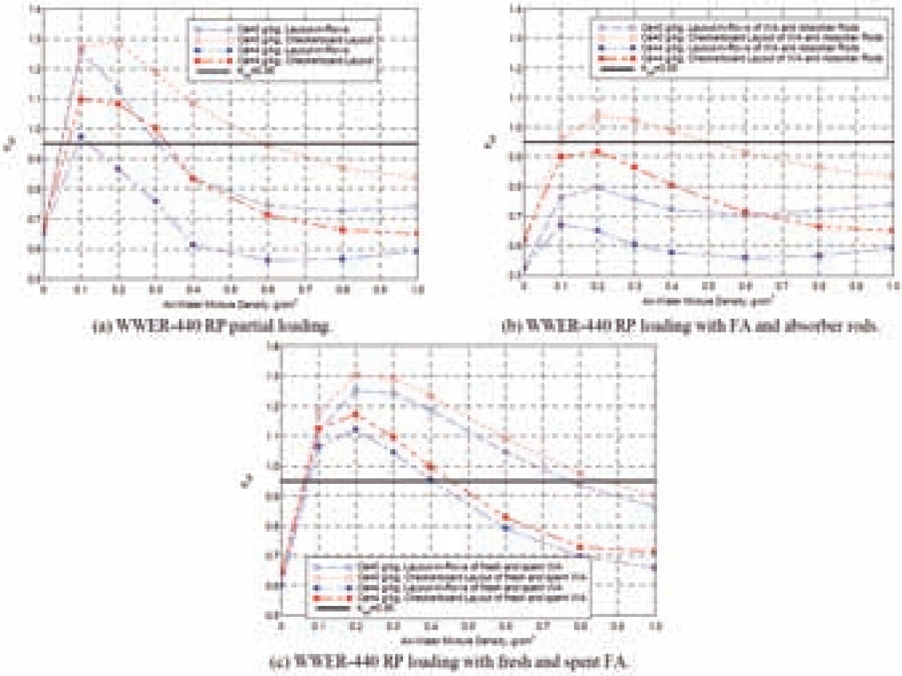


FIG. 6. k_{eff} dependence on air–water mixture density.

has been produced and the possibility exists for assessing the most likely ways of accident progression.

4. CONCLUSIONS

The reactor pool for existing fuel types of WWER-440 reactors meets nuclear criticality safety conditions under normal operation only when it is completely filled with borated water. The analysis shows that the conservative conditions required by regulatory documents leads to violations of the safety criterion over a wide range of changes in the air–water mixture density.

Based on the above calculations, the following can be concluded:

(a) The criticality of the reactor pool compartment is $k_{eff} = 0.5161 \pm 0.0006$ for normal operating conditions and the boric acid concentration at 16 g/kg (approximately 2800 ppm).

(b) The maximum k_{eff} for reactor pool racks under an optimum moderation (air–water density at 0.25 g/cm^3) is k_{eff} = 1.3394 ± 0.0007 (unborated water is credited).
(c) Taking credit for boric acid dissolved in the reactor pool water does not correct the situation. Even at C_B = 16 g/kg (approximately 2800 ppm), k_{eff} = 1.1752 ± 0.0004 (water density 0.117 g/ cm^3).
(d) In the analysis with simultaneous credit taken for boric acid and fuel burnup, the system reaches k_{eff} < 0.95 at the level of 50 GW d/t when the boric acid concentration is C_B > 8 g/kg (approximately 1400 ppm). But under such conditions, it is necessary to divide the reactor pool into individual zones — for high and low burnup (or fresh) fuel. Low density loading or an alternate loading with high burnup fuel, or addition of an absorber in the material of the rack elements could be ways for achieving the required level of criticality for a zone with low burnup fuel. This situation has resulted from the commissioning of new fuel types and fuel cycles. The WWER-440 systems were originally designed for fuel with 3.6% enrichment, but now this value is 4.4%. The hole in the fuel pellet centre has decreased and the amount of uranium in each assembly has increased as a result.

Now the most reliable solution for preventing the violation of regulations is partial loading of the spent nuclear fuel management system, followed by expert safety assessment. The inconsistencies between pool safety assessment results and regulatory requirements should not be considered as drawbacks of the system. Rather, they should be seen as the consequences of the obsolescence and excessive conservatism of existing Ukrainian regulatory requirements. The analysis allows the following recommendations to be made as regards the revision of the regulations to bring them into compliance with up to date international practice:

(1) Determine the minimal possible concentration of the boric acid dissolved in the pool water and conduct an analysis with boron credit. For example, such an approach is used in the USA.
(2) Determine individual racks or zones for separate location of fresh (slightly burned up) and spent nuclear fuel. Perform a criticality safety analysis of racks for fresh (slightly burned up) fuel and determine possible schemes for their loading. Conduct a burnup credit analysis of the racks for spent nuclear fuel. Such an approach is apparently being introduced in Germany, Spain and the USA.

The density of the moderator (air–water mixture) has a great influence on the reactivity of uranium–water systems. The criticality safety is usually (including this work) analysed for the whole range of possible changes in the density of this mixture (from 0 to 1 g/cm^3). This range should be reduced to more realistic values (moderator credit), but these values must be determined only by means of a very careful analysis (most likely by using probabilistic safety analysis). At present there is no information on the use of 'moderator credit' in the practices of a regulatory authority in any country.

The burnup credit approach is the most advanced and widely used in international practice. But all analyses and calculations related to its introduction pertain to PWR and BWR fuel. This approach formally complies with Ukrainian standards in force.

To introduce the burnup credit principle, as in advanced countries for PWRs and BWRs, the following is required:

(i) To develop and implement a methodology for determining fuel burnup (by instrument monitoring or calculation means);
(ii) To develop and implement a methodology for determining isotopic composition of spent fuel, depending on burnup;
(iii) To develop and implement a methodology for determining the neutron effective multiplication factor, depending on isotopic composition of spent nuclear fuel.

The following main conclusion can be made: the implementation (through the modification or updating of standards) of the approaches being used in many advanced countries for criticality safety analysis for the analysis of WWER spent fuel management systems will allow the elimination of most of the contradictions between regulatory requirements and the actual state of affairs.

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INTERNATIONAL SAFETY REGIME

(Session 3)

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LICENSING ASPECTS OF SPENT FUEL STORAGE AS A FUNCTION OF TIMESCALES

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Abstract

The paper is focused on the influence of different spent fuel storage time horizons on the regulatory approach and on the basic principles for regulatory decision making. Different national licensing strategies for planned and operating storage facilities, the role of related research and development projects and the results of safety assessments for important safety features are described. The paper includes a discussion on the limits and conditions that provide a practical framework for the safe operation of storage facilities.

1. INTRODUCTION

For countries relying on the supply of electricity from nuclear sources the management of spent fuel (SF) is an important subject. As the inventory of SF generated worldwide increases, the storage of SF is beginning to be planned for comparatively long periods. The reason for this development is that many nuclear countries do not have a clear national policy on SF management and follow the ‘wait and see’ approach. This approach allows the final decision on SF management to be postponed and allows the possibility of taking advantage of any important future technological developments in this area. On the other hand, it can slow down projects related to the siting, construction and operation of disposal facilities which are capable of accommodating the SF or HLW generated as a by-product of SF reprocessing. The disposal concept will also play an important role in relation to other options for future SF management such as partitioning and transmutation. The volumes and activities of HLW might change, but there will still be a need for their final placement in stable geological formations to isolate the waste from the biosphere.

A generally agreed definition of long term storage does not exist however. Typically, the period of concern covers a range from 30 to more than 100 years. Some regulatory authorities face challenges due to the fact that initially the SF storage facilities were considered as short term solutions. This is the case of several Eastern European countries, where it was expected that the SF would be sent for reprocessing to the former Soviet Union. Because of the political changes of the last two decades this option is no longer available under the original conditions and the SF storage period in existing facilities outside the Russian Federation has to be extended. Other regulatory issues are linked to the licensing of changes in the storage technology, such as those due to increases in storage capacity. A nuclear regulatory body may also be faced with other challenges during the licensing process when a new long term storage facility is planned or under construction.

All the above mentioned licensing aspects have to be clearly defined and assessed from the point of view of national regulations, taking into account the safety functions of specific storage technologies. The objective of this paper is to demonstrate the recommended licensing approach by which the regulatory body is able to maintain the safety of SF storage facilities for the operating period, taking into account the long term effects on SF and storage facility components. The licensing approach should also provide an interface to the next step in SF management, that is direct SF disposal or reprocessing. The scope of the paper covers the regulatory challenges in the process of issuing operational licenses for both wet and dry storage facilities for SF from power reactors in different timescales.

2. OVERVIEW OF STORAGE TECHNOLOGIES

There are several ways of categorizing the technologies used for the storage of SF. The safety functions of the storage technologies allow the regulatory body to assess the significance of the storage components used, and the issues related to the components of greatest safety significance will receive the regulatory body's priority attention. For the purpose of this paper, two storage design categories are considered — wet and dry storage facilities.

2.1. Inventory of spent fuel stored worldwide

The data presented is obtained from the IAEA's database of civilian nuclear fuel cycle facilities, the Nuclear Fuel Cycle Information System (NFCIS) [2]. NFCIS contains not only information on SF storage facilities, but also on facilities for uranium ore processing, recovery of uranium from phosphoric acid, uranium refining, conversion and enrichment, uranium and MOX fuel fabrication, etc. The facilities considered range from those at the planning stage to those at the decommissioning stage, so as to offer as complete an overview of the industry as possible.

The worldwide inventory of SF stored away from reactor (AFR) in both wet and dry storage is about 93 000 tonnes HM (31 December 2003), as reported in NFCIS (Table 1). The storage capacity of wet storage facilities exceeds the storage capacity in dry storage facilities by a factor of 2.7. This ratio is expected to change in the future in favour of the dry approach.

2.2. Design features of storage technologies

Wet storage technology is the oldest type of storage used for ‘at reactor’ (AR) and AFR and has been in use for more than 50 years. The benefits provided by this technology are mainly associated with effective cooling and shielding and direct control over stored SF. From the perspectives of long term storage, several disadvantages can be identified, such as the active nature of the storage method. It relies on permanent water quality control, involves the generation of radioactive waste (RAW), and can involve corrosion of the cladding due to direct contact of SF with water.

The use of dry storage technologies started in the early 1960s, and since this time they have evolved into a variety of systems. Examples of these are metallic or concrete thick wall containers (CASTOR, CONSTOR, TN) and canister-in-cask and canister-in-vault designs (NUHOMS, CASCAD). Dry storage of SF in an inert atmosphere, which is used in most dry storage systems, provides favourable conditions for long term storage. However, the higher cladding temperatures and the lack of any possibility of having direct control of SF conditions are considered to be disadvantages of this technology.

TABLE 1. CURRENT AND PLANNED INVENTORY OF SF STORED AT AFR STORAGE FACILITIES, AS REPORTED IN NFCIS [2]

AFR storage type	In operation	Constr.	Awaiting license	Planned	Shut-down	Decomm.	Standby	Other	Total
Dry SF storage (t HM)	25 752	12 889	140	123 173	N/A	N/A	35	1370	163 359
Wet SF storage (t HM)	67 110	N/A	N/A	N/A	55	1870	750	N/A	69 785
Total	92 862	-	-	-	-	-	785	-	233 144

3. SAFETY FEATURES OF STORAGE TECHNOLOGIES

The regulatory body has to be technically competent and capable of assessing the safety relevant features of the storage system as a part of the licensing procedure. The first step in the regulatory decision making process, which encompasses the facility licensing process, is a clear definition of issues and the assessment of their safety significance [1]. For storage of SF these steps can be transformed into the identification of safety features of licensed facilities.

The safety features of SF storage facilities can be divided into two main groups:

(1) Features of fuel assemblies: Intact cladding during storage represents the primary barrier providing the containment of radionuclides and allows the safe retrievability of SF in the subsequent SF management steps (such as transport, reprocessing, direct disposal, etc.), or in the case of abnormal events. In the case of damaged fuel the cladding functions can be replaced by a hermetically tight case. The definition of damaged fuel is country

TABLE 2. OVERVIEW OF SELECTED FEATURES OF DRY AND WET STORAGE TECHNOLOGIES

Wet storage technology		Dry storage technology	
Advantages	Disadvantages	Advantages	Disadvantages
Worldwide long term experience	Active storage technology relying on permanent water quality control	Passive storage technology	Higher cladding temperatures
Effective shielding and heat removal properties of water	Direct contact of SF with water	No or minimum amount of RAW generated during normal operation	Direct control of stored SF not possible
Direct control of stored SF	Generation of RAW during normal operation	Favourable conditions for long term storage of SF in inert gas environment	
	Risk of loss of shielding and cooling media (water)		

specific, ranging from 'mechanically damaged but tight' to 'leaky fuel assemblies'. The classification of fuel according to its cladding integrity is performed by the 'sipping test' in reactor pools before transport to wet storage or before loading to dry containers/canisters. For dry storage technology, the SF is exposed to additional thermal loads during the container/canister vacuum drying procedure when, for a short time period, the cladding temperature can significantly exceed 350–400°C and can lead to additional cladding damage.

(2) Features of storage technology: These safety features are identical for wet and dry storage facilities, even if they are achieved by different technical means. They cover criticality control, removal of decay heat, radiological protection, retrievability and containment of the radioactive content of the fuel.

For wet storage pools, water provides cooling and γ and neutron shielding, the storage facility itself provides containment and criticality is prevented by spacing, use of boron materials and/or boron additives in water. The chemical properties of water (such as concentration of chlorine ions), its temperature and the presence of corrosion products in water can have a significant impact on the long term performance of storage pools.

For dry storage, the safety features depend on design and on the construction materials used. For thick containers whose walls provide all the safety functions, the aging mechanisms (of metallic materials and concrete) play the crucial role. For canister-in-cask and canister-in-vault technology, the safety functions are distributed between the thin wall canister (criticality control, heat removal and retrievability) and the concrete cask or vault (shielding, protection from mechanical load and heat removal).

After the identification and classification of safety issues the regulatory body has to identify the legal criteria governing its decision. This element of the regulatory decision making process lies outside the scope of this paper. However, the next step, which includes the collection of all relevant data and information from operational experience, the results of research and development projects, and safety assessments, is elaborated in the following sections of the paper. Additionally, for complex situations which may occur during the storage of SF, the regulatory body may need to request or to support new research projects as an input to its decision making process.

3.1. Degradation mechanisms of fuel cladding

Cladding surrounding the fuel pellets is the first mechanical barrier that protects the fuel if it is not damaged and prevents release of the radioactive inventory. In relation to the disposal of SF, cladding can be assumed as one of the barriers separating fuel from components of the engineered barrier system and the host rock. Figure 1 shows the containment provided by the cladding of commercial spent fuel after its disposal. Here the cladding makes a difference of about two orders of magnitude at between about 10 000 and 40 000 years after SF emplacement in the repository. As the cladding fails, the two dose curves become similar after about 70 000 years [3].

The cladding degradation mechanisms that may affect the cladding integrity during wet storage are as follows [4]:

(a) Mechanical stress as a result of internal gas pressure (He and fission gases).
(b) Corrosion (uniform, pitting, galvanic...). For Zr cladding under controlled wet storage conditions, corrosion is up to 1010 times lower than under reactor operating conditions at about 10^{-6} mm/a [4]. Studies show that after the initial growth of an oxide layer of up to several microns in the first years of wet storage, no additional corrosion is expected to occur (at least for up to several decades).
(c) Hydriding, i.e. absorption of hydrogen released by the reaction of Zr with water, can influence cladding integrity related properties such as yield stress, rupture stress and uniform elongation. SF during reactor service can accumulate up to 600 ppm of hydrogen in the cladding, but under wet storage conditions, when the temperature is low, hydrogen redistribution by thermal diffusion can be ruled out (unlike under dry storage conditions).

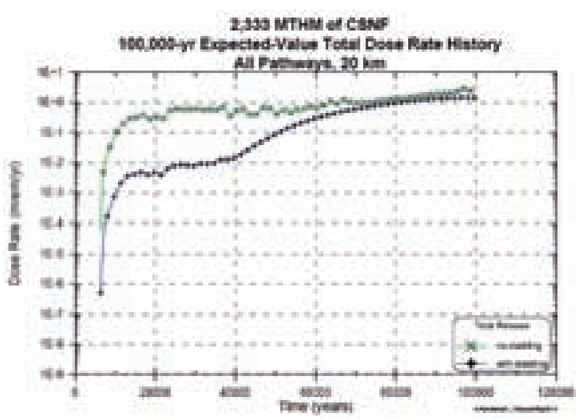

FIG. 1. Impact of Zircaloy cladding on the total performance of geological disposal [3].

For dry storage conditions, creep is generally considered to be a limiting degradation mechanism. As long as cladding hoop stress remains below the cladding yield strength, creep rupture is very unlikely.

Example 1: The cladding deformation was assessed for different cladding materials in several studies. Reference [5] contains the results of simulated cladding behaviour for Zr-1%Nb alloy used in WWER and RBMK fuel cladding. The results for dry storage conditions in the CASTOR 440/84 container show (see Fig. 2.) that the hoop strain is at 0.025%, comfortably below the total hoop strain limit of 1% and after about 15 years of simulated storage conditions, when the cladding temperature has dropped well below 250°C, it remains practically unchanged.

Example 2: This example comes from a US survey of spent fuel status after 15 years of dry storage in CASTOR V/21 containers. Fuel from the Surry NPP with burnup of about 36 GWd/t U showed a maximum creep of less than 0.6%, no additional release of fission gas during the storage period and no evidence of hydrogen pickup [6].

Other potential mechanisms that may affect the cladding integrity are cladding oxidation (storage under inert gas conditions rules out the presence of oxidising substances), crack propagation and hydrogen induced defects such as delayed hydride cracking.

3.2. Degradation mechanisms of storage technology

Another set of degradation processes potentially affecting the safety functions of storage facilities is related to the components and subsystems of

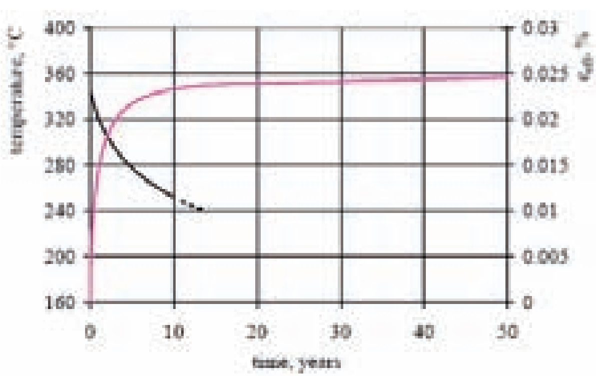

FIG. 2. SF storage performance prediction for a WWER-440 fuel rod [5].

this technology. For wet storage facilities, corrosion rates for some materials used in wet storage systems can become an operational life limiting factor. Corrosion resistance of construction materials ranges from quite high (stainless steel) to poor (carbon steel, cast iron). Corrosion can also influence the performance of concrete structures.

Radiation effects on storage racks, pool liners and water chemistry may lead to other processes, such as the generation of such long lived species as H_2O_2 and an increase in the oxidizing potential of water and degradation of neutron absorbers based on silicon (if used).

For dry storage facilities, the components expected to require attention are sealing systems, transport interfaces (e.g. trunnions), neutron moderators, monitoring equipment, welds (if used) and concrete (if used). The inert or nitrogen atmosphere inside containers or canisters minimizes the corrosion of not only SF, but also construction materials. However, these favourable conditions can change if seals fail and the storage system components are exposed to corrosion in air.

Example 3: In its research and development programme on dry storage technology the Japanese research institute CRIEPI is performing several tests of the long term durability of components of metallic and concrete containers. One of them is a project to investigate the long term integrity of metallic container seals [7]. Under accelerated conditions, when the sealing temperature remains constant, storage periods of 50 and 80 years were simulated (after 15 years of actual test duration). The leak rates for aluminium sealing remained almost unchanged at the value of 10^{-9} Pa·m^3/s (see Fig. 3).

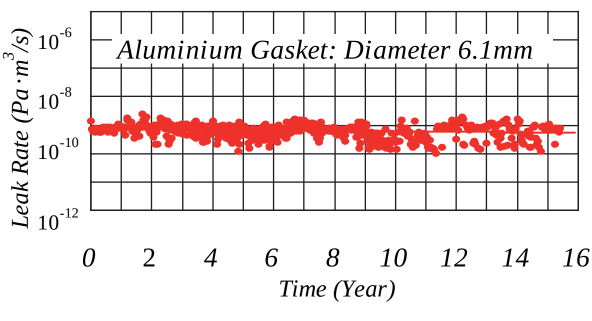

FIG. 3. Leak rate of aluminium sealing during long term testing [7].

4. LICENSING STRATEGIES

After the regulatory body identifies the safety features and related issues as part of its integrated decision making process, it has to collect all relevant data for making its decision, to obtain the views of internal or external experts and then, based on regulatory criteria, make its decision — to issue or reject the request to issue the licence.

For wet storage facilities and for some dry storage technologies such as canister-in-vault storage facilities, the licence covers the whole storage system. For container storage facilities, where the building of the storage facility has no or limited safety functions, separate licensing of the thick wall container, which provides all safety functions, is required. The licensing strategies for these containers can follow the same strategies as for storage facilities.

For facilities with operating periods of several decades there are three basic licensing strategies:

(1) Permanent licence issued for the entire anticipated operating period (used mainly in Western European countries, Japan, the Republic of Korea, etc.); the conditions of the permanent licence usually contain requirements for periodic safety review.
(2) Limited licence issued for time periods of no more than 10–20 years (used mainly in Eastern European countries); the facility's performance is reassessed periodically and if conditions for safe storage are fulfilled the licence is renewed.
(3) Licence for an extended operating period and for a changed storage facility design. The need for extension of the storage period beyond the expected design lifetime can result from, for example, a changed SF management strategy, such as when SF originally expected to be reprocessed after short term storage has to remain in storage for a much longer time because the reprocessing option is no longer available.

In general, there are three basic initial reasons for facility design change:

(i) Changed functionality of the facility, such as extension of storage capacity, use of different container types and storage of SF with higher burnup;
(ii) Safety issues raised during facility operation;
(iii) Changed national regulatory framework, international recommendations, etc.

National regulatory systems should contain the formal administrative tools to resolve such issues. They should: clearly define the situations when the licensing process has to be initiated; contain requirements on safety case structure; and require detailed descriptions and justifications of facility modifications and details on the safety assessment, etc.

5. LIMITS AND CONDITIONS FOR SAFE LONG TERM STORAGE OF SPENT FUEL

As part of its licensing procedure the regulatory body has to have in place a set of operational limits and conditions (OLCs) that the operator of an SF storage facility must follow in order to ensure safe operation of the facility. The safety case defines the OLCs under which spent fuel elements are stored, including e.g. in-store environmental conditions. The OLCs can be divided into three main sets, safety specific, technical and organizational.

Safety specific OLCs are directly related to the safety function of the components of storage technology which ensure protection of workers and the public from undue radiation exposure. The components requiring special attention are listed in Table 3.

Safety specific OLCs cover the maintenance of:

(a) Subcriticality, which is linked to both the short term and long term effects which may occur during pool or container/canister loading and transport and, over longer time horizons, to the reconfiguration of SF as a result of material degradation or accidents, degradation of neutron absorbing materials incorporated into the storage technology, etc. Under normal operating conditions the increase of reactivity is very unlikely and the regulatory limits for k_{eff} should not be exceeded. For facilities in which burnup credit is being utilized the use of this approach must not have negative consequences on the facility's safety. Therefore, the regulatory body has to require the independent verification of the actual SF burnup

TABLE 3. EXAMPLE OF STORAGE COMPONENTS REQUIRING SPECIAL ATTENTION

Wet storage	Dry storage
Water (composition)	Sealing system
Pool liners	Container/canister handling system
Neutron absorbers	Neutron moderator
Storage rack	Monitoring equipment

(by measurement) of each loaded assembly to make sure that the real burnup has reached the threshold determined by the calculations.

(b) Fuel integrity, which is related to all safety functions of the storage facility. The short term effect related to the transition from wet conditions in the reactor pool to transport and storage conditions may lead to the time limited increase of cladding temperature. In the case of dry storage the vacuum drying process of the container/canister shaft with loaded SF can lead to a significant overrun of the cladding temperature limits, which for zirconium based cladding usually lie in the range of 350º–380°C.

The calculated cladding temperature for different thermal loads of the CASTOR 440/84M container, performed as a part of the container safety case, show (see Fig. 4) that the drying process does not need to be limited by time if the thermal output of the entire container inventory does not exceed 17.1 kW. For thermal outputs higher than this value the drying process has to be interrupted after about 25 to 70 hours, depending on the thermal output, and the conditions for heat removal have to be re-established [8].

The monitoring of gases during the container or canister drying process can be used to determine the integrity of SF cladding by measuring the activity of noble gases (Xe-133, Kr-85) and aerosols (Cs-137, Co-60).

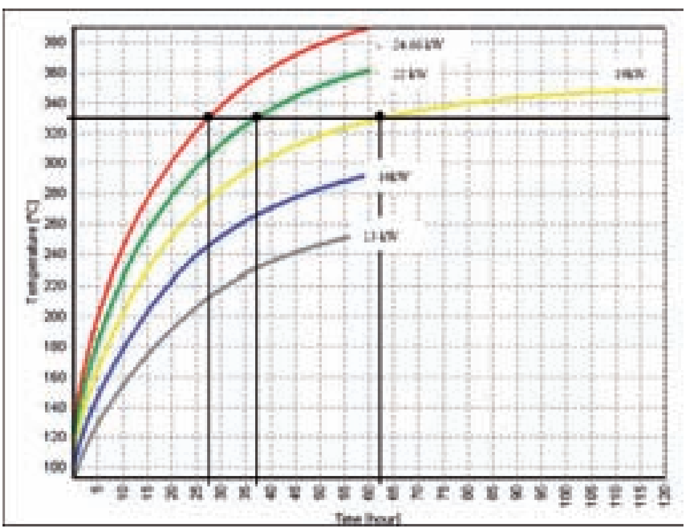

FIG. 4. Calculated cladding temperature during vacuum drying of a CASTOR 440/84M container [8].

Long term effects on integrity (covered in Section 3) are related to the creep, corrosion, hydrating and other chemical and physical processes. The assessment of these effects leads to the definition of other conditions contributing to the long term integrity of fuel cladding, such as specifications of water properties in the case of wet storage facilities (temperature, pH, conductivity, chemical composition, specific radioactivity).

Retrievability in general refers to the ability of SF to be removed from the storage facility. For dry container/canister storage facilities it concerns both the retrievability of fuel assemblies from the container/canister and the retrievability of the container/canister itself. The storage concept must allow the recovery of all SF at any time during storage, at the very latest before the disposal or further SF processing. The retrievability is ensured by:

(1) Technical conditions such as availability, maintenance and upgrading of standard manipulation and handling equipment (cranes, manipulation tools) and of tools to handle canned assemblies and provisions for short and long term retrievability (reserve storage capacity, hot cells);
(2) Organizational conditions such as operating procedures for periodic inspections and maintenance of handling equipment and its components.

Another set of safety specific OLCs is related to the lifetime of safety relevant storage components. These components and their physical and chemical parameters differ for wet and dry storage, but both have to be the subject of continuous long term maintenance, testing and surveillance programmes, and inspections and ageing management.

As mentioned in Section 3, the regulatory body has to assemble all available information which can contribute to its decision on facility licensing. One source of this information are the outputs from different continuous programmes at the facility. These programmes provide evidence that a storage facility and its components operate and will operate as expected over the lifetime of the facility. They cover items such as the periodic evaluation of the facility's performance and the reporting of corrective actions, including the replacement of components with limited service life, etc.

Ageing management programmes should identify:

(i) Components with limited lifetimes and components designed for the entire expected operating period;
(ii) All ageing mechanisms important to safety related components (and determine their possible consequences and the necessary activities in order to maintain their operability and reliability).

TABLE 4. EXAMPLES OF OLCs FOR WET AND DRY STORAGE

Wet storage		Dry storage	
Safety specific OLCs	Storage pool inventory (no. of assemblies, total activity, thermal output, initial enrichment, cooling time in reactor pool, burnup)	Safety specific OLCs	Total inventory of the facility (no. of containers/canisters)
	Integrity of SF assemblies (intact and canned fuel)		Container/canister inventory (no. of assemblies, total activity, thermal output, initial enrichment, cooling time in reactor pool, burnup)
	Reserve storage capacity		Integrity of SF assemblies (intact and canned fuel)
	Limits of cladding temperature during manipulation and transport		Reserve storage capacity
	Water properties, including conditions for use of n-moderators (if applicable)		Conditions for container/ canister drying
	Water treatment and cooling system		Limits of cladding temperature during manipulation, transport and storage
	Decontamination and dosimetric control of transport container		Decontamination and dosimetric control
	Maintenance, testing, surveillance programmes and inspections		Maintenance, testing, surveillance programmes and inspections
	Ageing management programme		Ageing management programme
Technical OLCs	Max. weight of fuel assemblies	Technical OLCs	Max. weight of loaded container/canister
	Safeguards system		Container/canister surface temperature
	Physical protection		Safeguards system
	Radiation monitoring system		Physical protection

TABLE 4. EXAMPLES OF OLCs FOR WET AND DRY STORAGE (cont.)

Wet storage		Dry storage	
Admin. OLCs	Record management system		Radiation monitoring system
		Admin. OLCs	Marking of container/canister Record management system

The results and conclusions of ageing management programmes should be considered in periodic safety reviews whose role is, among other things, to assess ageing mechanisms and to determine whether they have been correctly taken into account. Technical OLCs result from the design of storage technology and characterize its function and way of use. These conditions are not directly related to the safety functions of the storage facility, but may influence them or may be related to the conventional (non-nuclear) safety functions. In addition to the technical conditions mentioned in connection with SF retrievability, these OLCs cover items such as the total weight of the container/cask, the use of protective covers over pools (for wet storage), the container/cask surface temperature, the use of the nuclear safeguards system (seals, cameras), physical protection, etc.

The last set of OLCs are the administrative limits and conditions covering the requirements on container/cask marking, the reporting of operational accidents, the record management programme, etc. The long term record management programme requires information on the location and characteristics of every SF element in storage, including information on its ownership. Records in the system have to be preserved and updated (taking into account, in particular, the condition of SF during storage), to enable implementation of the strategy for the future SF management, including disposal. Issues related to the long term record management are addressed by a number of national and international projects [9].

6. CONCLUSIONS

Extension of SF storage timescales creates new challenges for national regulatory bodies. These challenges can be addressed by:

(a) The clear definition of regulatory requirements and guidelines including, not only quantitative criteria for safe long term SF storage related to the safety functions of storage components, but also programmes of maintenance, testing, surveillance and inspections and ageing management programmes.

(b) The periodic reviews of the safety case for storage facilities, including containers or canisters (if used), focusing on the performance of the facility during the last licensing cycle and the expected future performance. Periodic safety case reviews contribute to the updating and revision of OLCs in the light of operational experience and developments in science and technology.

(c) The support of research and development programmes in building the scientific background for the regulatory decision making processes, including the licensing of SF storage facilities.

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PANEL DISCUSSION

(Session 3)

TERM RELATED LICENSING ISSUES

Chairperson: **W. BRACH** (USA)

Members: **A. Lavrinovich** (Russian Federation)
V. Khotylev (Canada)
K. Shirai (Japan)
P. Standring (United Kingdom)
I. Pechera (Ukraine)

STATEMENT

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In the past year we have renewed two dry cask storage licences for terms of an additional 40 years. The licences were originally issued for a 20 year term. The renewals were very challenging, both from a technical and from a policy perspective, and also it was a first of the kind licensing action for the Nuclear Regulatory Commission (NRC). The requests of the licensees necessitated a Commission level policy review and determination. Our staff felt, from a technical standpoint, that there was adequate technical information available to support the approval of a 40 year term for the licence extension. Draft guidance had been developed to assist the NRC's review and the licensees' review of the technical and programme issues associated with licence renewal. This document, entitled Preliminary NRC Staff Guidance for Part 72 License Renewal, was issued in March 2001 and is publicly available through the NRC web page. Guidance on materials behaviour, also related to compliance with safety criteria for the storage of spent nuclear fuel, is given in the American Society for Testing and Materials (ASTM) Standard Guide C 1562. It is entitled Standard Guide for Evaluation of Materials used in Extending Service of Interim Spent Nuclear Fuel or Dry Storage Systems. These two guidance documents guided the review.

In our renewal process it was extremely important to identify and list all of the components of the dry cask storage system, to categorize those components in order to determine which are important to safety, analyse the important to safety components in order to ensure that each would perform its intended safety function during the period of licence renewal, and determine appropriate and relevant surveillance and maintenance needs for each of those important to safety components. Other aspects of the review focused on environmental considerations, management controls, institutional controls, quality assurance and security.

STATEMENT

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A common problem is concerned with the capacity of storage facilities becoming exhausted, especially at reactor sites. For this reason it was decided 20 years ago to build a special interregional storage facility in the Russian Federation. More recently, it was decided to build a new dry storage facility initially for 50 years of operation.

The national strategy for spent fuel storage depends on the national policy for spent fuel management, and the time related issues depend on this. In the Russian Federation we reprocess some kinds of spent fuel, but not all; for example, RBMK-1000 spent fuel, is not reprocessed while other types of spent fuel, for example WWER-1000 spent fuel, are intended for reprocessing and we are constructing a facility for reprocessing such fuel.

The management of spent fuel that is not being reprocessed is a common problem, and in many countries decisions on the subject have been postponed. As a result spent fuel is in intermediate storage.

We have had problems with aluminium clad fuel stored for many decades in water pools where corrosion has occurred, and it has therefore been decided to reprocess the fuel in the near future.

STATEMENT

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Introduction

Irradiated fuel unloaded from nuclear reactors in Canada is mostly natural uranium fuel. This fuel is typically unloaded during on-power refuelling. The fuel is relatively low burnup and, consequently, has low decay heat. Another aspect is that there are no plans in Canada to reprocess and recycle this fuel. So current plans are based on direct long term management of the spent fuel.

Term categories of spent fuel management

There are three term categories of spent fuel management at Canadian nuclear power plants. These categories are

(a) Short term storage in the fuel bays (pools) at the nuclear power plants;
(b) Medium term storage at dry storage facilities;
(c) Long term storage.

Term related licensing aspects of short term waste management

The relatively low fuel burnup and, consequently, the low decay heat influence the term related licensing issues. Most of the term related issues raised by the Canadian Nuclear Safety Commission (CNSC) are in the areas of radiation doses to workers from normal operation, malfunctions and accidents, acceptable thermal loads, and acceptable characterization of decay heat.

Term related licensing aspects of intermediate waste management

Term related issues are routinely raised by the CNSC during various stages of the licensing process for dry storage facilities, including the preliminary safety report, the environmental assessment study, the construction approval and the operating approval.

Many technical licensing issues are raised by the regulatory body staff during their review of documents on the storage container (SC). Criticality is usually one issue which takes considerable effort to resolve. There are typically a large number of technical licensing issues on the thermal analysis of the storage container. The analysis of the effect of air gaps between concrete and carbon steel liners (inner and outer) is an example of an issue to be analysed to satisfy the regulatory body. However, the key term related requirements are those for acceptable thermal loads such as ambient temperature in the storage facility and fuel cladding temperature in the SC cavity.

The typical SC is designed for a minimum 50 year life but will likely last much longer. However, research programmes carried out by the industry have assessed the behaviour of spent fuel when stored in dry and moist air conditions, and in a helium environment. It was concluded that CANDU fuel bundles, whether intact or with defects, can be stored in dry conditions for up to 100 years or more without losing integrity.

In the context of discussions on the term related licensing issues it is appropriate to mention that the licence period (i.e. validity of a current licence) is much shorter than the design life of the SC. Amongst other things, the short duration of the licence allows a flexibility to allow incorporation into the licensing documentation of an appropriate ageing management strategy and specific measures. Thus the primary licensing issues are related to ageing management, surveillance and maintenance.

Licensing aspects of long term waste management

Licensing aspects of long term management arise from Canada’s national policy on the long term management of nuclear fuel waste.

The Nuclear Fuel Waste Act implements a key component of the Government of Canada’s 1996 Policy Framework for Radioactive Waste — that the Federal Government, through effective oversight, would ensure that the long term management of radioactive waste is carried out in a comprehensive, integrated and economically sound manner. The key elements of the Act include:

(a) Requiring the major owners of nuclear fuel waste to establish a waste management organization to carry out the managerial, financial and operational activities to implement the long term management of nuclear fuel waste;
(b) Requiring the major owners of nuclear fuel waste to establish trust funds and to make annual payments into those trust funds to finance the long term management of nuclear fuel waste;

(c) Authorizing the Governor in Council to make a decision on the choice of approach for long term management of nuclear fuel waste for Canada to be implemented by the Nuclear Waste Management Organization (NWMO).

Recently, the NWMO submitted its proposed options for long term management of nuclear fuel waste to the Federal Government, which will have to choose from the proposed options, including deep geological disposal in the Canadian Shield; centralized storage either above or below ground; and surface storage at nuclear reactor sites. That decision will reveal the long term strategy. The term related licensing aspects and requirements for nuclear fuel waste storage will be delineated within the context of that strategy.

STATEMENT

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Last November an agreement relating to the construction of an away-from-reactor storage facility was concluded in my country between local governments and utilities with a possible start of commercial operation of the facility by 2010. Based on this experience I will give some key points relating to long term storage.

Under the agreement, the maximum storage period will be 50 years. This is to remove any suspicions in the minds of the public that the storage might be permanent.

The licence application for safety design approval will be submitted by the storage facility development company. The competent Japanese authorities issued the safety related technical requirements for an interim storage facility using dry metal casks and concrete casks in April 2006.

From this document the key safety requirements relating to long term storage can be determined. The first item is the degradation of components important for safety, especially the long term integrity of the confinement. In the case of the metal casks, their integrity should be maintained under high temperature conditions at the beginning of storage and in a corrosive environment after a few decades of fuel storage. Moreover, during the subsequent transport, the containment safety of the metal casket must be well explained so as to obtain public acceptance for construction.

Another degradation issue discussed seriously in Japan is the atmospheric stress corrosion on the welded canister surface. As the chloride amounts transported by the natural cooling air from the sea cannot be neglected for long term storage, lifetime evaluation and periodic inspections will be needed even if stainless steel is used.

The next item is characteristic of the types of spent fuel expected in the near future. Higher burnup fuel will be discharged and transported for storage. Methods for the evaluation of the integrity of high burnup spent fuel in dry storage and subsequent transport and a source term for those evaluations have to be determined.

Another key item for Japan is seismicity. Amended seismic safety guidelines will be enforced next August. Seismic safety must be guaranteed for the spent fuel storage container, the storage facility and the corresponding equipment for long term storage.

Some of these technical matters should also be key issues for other IAEA Member States, and I hope that they can be solved through cooperative international activities like this conference.

STATEMENT

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THE CONTINUED USE OF EXISTING SPENT FUEL STORAGE FACILITIES

Re-licensing the continued operation of existing spent fuel storage facilities is primarily a matter of substantiating the existing safety related systems, structures and components (SSCs) against the effects of ageing and the new duty requirements.

The act of re-licensing or licensing existing facilities for change of use can be viewed as an opportunity to address wider stakeholder issues and regulations that have been introduced since the facilities were designed and built. The process, however, can introduce a number of interesting dilemmas created by the way the industry has evolved with time.

One example is concerned with the possible influence of 'new build' requirements on re-licensing. Examples of these are seismic qualification, severe wind impact, aircraft impact, building design codes or changes in methodology used to undertake safety assessments. If the facility was designed in the early 1980s, it is likely that the building design codes will have changed. At that time there would not have been a requirement to build the facility to a specific seismic standard and if current regulations were rigorously applied then, in principle such facilities would have to be replaced. However, by means of assessment or modification, the SSCs may be shown to still be 'fit for purpose'.

To date, redundancy in design has enabled many existing storage facilities to continue operating, but as we look to increasing storage durations from the current 20–40 years to 50–100 or even 150 years the question arises as to whether evolving safety case methodologies and design standards will present a problem for continued operations in the future.

Design standards is not the only area where there have been changes. Since the 1990s there has been much greater emphasis on environmental issues. For most existing storage facilities there was probably no requirement to justify operations in terms of environmental impacts at the time of their commissioning. Any consideration of an environmental impact at that time would have been focused on impacts local to the facility; the emphasis being on meeting radiological protection requirements. This is no longer the case. For new building projects in a number of countries it is required that the wider environmental issues be assessed to show that it is the best practical option (BPO) or best practical environmental option (BPEO) and then it must be demonstrated that the operations are undertaken using best practical means (BPM).

STAKEHOLDER INVOLVEMENT

In this context the situation has moved from 'reactive' to 'proactive'. Key stakeholders have their own views and perceptions of what is being done and they are heavily influenced by media activities. Nowadays it has to be recognized that the stakeholders have to be involved and in some cases they may need to be educated about planned activities. For the success of future operations they need to support the developments.

SOCIO-ECONOMIC IMPACTS

This question may seem of little relevance to a fixed term licence renewal, but when it is linked to a transition in site activities, for example from a production environment in support of power generation to a storage environment as a step to final spent fuel disposal, then the impact on the local and wider communities has to be carefully managed.

FUTURE CHALLENGES

The increased security threat from terrorist activities has taken the industry's awareness to a new level and it represents an area that will come under special attention in any re-licensing activities.

STATEMENT

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In many countries, licences are issued for every life stage of the spent fuel management facility — siting, construction, commissioning, operation and decommissioning.

One of the key parts of a licence is the term of its validity. It may be the period over which the assumptions underlying the licensing decision are judged to remain valid. At the end of the period the basis for licensing should be re-examined in order to extend the licence.

At present, the main facilities for spent fuel management are interim spent nuclear fuel storage facilities. A specific feature of such facilities is the concern with their decommissioning. It would only be possible to stop the operation of such facilities by transferring the spent fuel to another storage facility or to a reprocessing or disposal facility. In the first of these cases it is necessary to create a new storage facility and to provide for the transfer of the spent fuel to it. The creation of a new storage facility can be justified only when the safety of the existing storage facility cannot be guaranteed. A certain time period is needed for designing, licensing and constructing a new storage facility. Recognition that it too will have to be decommissioned should be taken into account in its design.

To terminate the operation of a spent fuel storage facility it is necessary to carry out a large number of activities, and experience shows that the necessary time period for their implementation could be longer than originally planned. This in turn requires the prolongation of the operation of existing facilities, and the safety of the facilities must be ensured for the extended time period.

The life extension of spent fuel management facilities is an increasingly important issue affecting many countries.

Panel Discussion

TERM RELATED LICENSING ISSUES

W. BRACH (Chairman — United States of America): I would now like to open the subject of 'term related licensing issues' for general discussion.

R. EINZIGER (United States of America): To date, only one storage canister has been opened up and the fuel looked at in the USA, and that was by way of an afterthought as part of another experiment, so that the baseline for the evaluation of the fuel was not what it could have been. With high burnup fuel, changes such as rim effects in the fuel and hydrided cladding may be expected. Are there any plans in any country to have a demonstration cask loaded with high burnup fuel, possibly with some cladding defects, so that at some time in the future, when it is time to extend the licence, there will be some data that would provide a baseline for scientific predictions?

K. SHIRAI (Japan): In Session 5 I shall be presenting a paper entitled Demonstration Test Programme using Full Scale Metal Casks and Concrete Casks, which may be relevant to that question.

I. PECHERA (Ukraine): In Ukraine we share such concerns about fuel behaviour in the future. That is why one of the conditions for licensing the operations of our storage facility is to carry out scientific research on dry spent fuel storage in order to provide guidance on how to correct any problems at the storage facility and on the time period for storage. My company, Energoatom, has a contract with the Russian Reactor Research Institute NIIAR in Dimitrovgrad, which is doing such research.

T. SAEGUSA (Japan): The Tokyo Electric Power Company has over ten years of experience of metal cask storage. Our Nuclear Safety Commission asked the company to open a cask every five years in order to inspect the integrity of the spent fuel. So a visual inspection, under water, of the spent fuel was carried out five years and ten years after the start of storage, but the burnup was not very high — about 30 GWd/t.

Z. LOVASIC (IAEA): In Canada where I come from several such experiments have been carried out, starting about 20 years ago at the Whiteshell Nuclear Laboratories. The fuel was taken out of dry storage containers on several occasions — the latest was after 18 or so years. Extensive tests carried out at the laboratories showed that at 18 years from the start of storage there were no detrimental effects of ageing on the fuel. Nevertheless, that is not enough and the project is continuing.

J. WHANG (Republic of Korea): I have two questions for Mr. Khotylev. In Canada, for how long are you going to store spent fuel in concrete casks? And do you plan to have periodic safety reviews of the concrete casks?

V. KHOTYLEV (Canada): Regarding the second question, an important point is that, although the design life of the dry storage facility is 50 years (and there is evidence that it could be 100 years), our licensees are licensed for five years. So if nothing changes in the regulatory regime in Canada then the licensees have to come back every five years.

Z. LOVASIC (IAEA): There are plans in Canada to continue investigating spent fuel integrity. Among other things there are long term plans to open storage containers after some 30 years. I would note that the experimental conditions at Whiteshell Nuclear Laboratories are more stringent and more aggressive than in a real storage facility.

W. BRACH (United States of America — Chairperson): Let us turn to the question of ageing. In our review of the two facilities which we relicensed last year, at the Surrey nuclear power plant and at the H.B. Robinson nuclear power plant, ageing management was an important element of the programmes that the licensees had in place to conduct monitoring and surveillance of the dry cask storage systems. This was important from our perspective and provided data which helped us in our decision to relicense those facilities for an additional 40 years.

V. KHOTYLEV (Canada): Ageing management is very important in view of the uncertainties, because there is no deadline yet for long term storage and we do not know exactly when it will be set, what the public reaction will be, what the decision of the government will be and when this work will be completed. Because we do not yet know any of these things we have to rely on ageing management, and so it is a very important activity. On almost a weekly basis we have various licensing requests or discussions related to ageing management.

V. CHRAPCIAK (Slovakia): In our unit we use Russian fuel. The oldest fuel in our wet interim storage facility is more than 20 years old, and up to now we have had only good fuel without failures. At the present time an inspection stand for checking the fuel is under construction in our interim spent fuel storage facility. In future we will check the fuel, but up to now we have had only good experience with it.

P. STANDRING (United Kingdom): A number of years ago we changed from probabilistic to deterministic safety cases. One of the challenges we faced when this happened was that, because of the new approach and the new safety criteria that were generated, it was no longer possible to do the same operation which had been done for the previous 20 years, or even longer. I would be interested to know if anyone else has experienced similar problems.

W. BRACH (United States of America — Chairperson): Some panellists identified the need for public engagement and public awareness with regard to some industry and regulatory plans and activities. Are there any further comments on this issue?

P. STANDRING (United Kingdom): This has become an important area for us in recent times because we are going through a transition from being an operating company that had contracts which were managed as a business to a company operating under the guidance of the Government. One aspect of the change is that the Government wishes to involve the public in what happens to the fuel in the future.

There is an ongoing dialogue among the key stakeholders. It is a fairly new process, certainly for the new body that has been created — the Nuclear Decommissioning Authority (NDA). It is taking place at a number of levels: there is a site stakeholder group which reflects the views of the communities around the site and there is a national stakeholder group which has sub-groups looking at particular issues.

The company always had ongoing dialogue with key stakeholders in the local communities. What has changed during the past 5–6 years is that the dialogue has become more national.

V. KHOTYLEV (Canada): Public involvement is a very important part of the Canadian licensing process. The widest public involvement we observe is at the stage of the so-called environmental assessment, which is done in accordance with the Canadian Environmental Assessment Act. The Canadian Nuclear Safety Commission engages in discussions with the public, and this early engagement is important for later steps in the licensing process because whatever concerns the public expresses at the early licensing stages are usually encountered at the later stages. Therefore, if we take account of those concerns at the early stages it helps later.

J.S. LEE (IAEA): There seems to be a wide variation in the degree to which storage facilities are protected. For example, in the USA there is usually no building covering the storage cask, whereas in Europe most facilities are covered by a building.

W. BRACH (United States of America — Chairperson): From the NRC's perspective the review of a storage facility is based on the inherent safety that is demonstrated by the proposed dry cask storage system. As you mentioned, in the USA most of the spent fuel storage facilities are located out of doors, not in a building or below ground. In my view, there is not necessarily a right or wrong approach; the review is based on the safety as represented/demonstrated in the application. There are also public acceptance aspects — facilities covered by a protective building or in some other way are not as easily seen from distant roads.

J.S. LEE (IAEA): I have a question for Mr. Marvy. The French Atomic Energy Commission (CEA) has built or is building a demonstration facility for very long term underground storage at the CECER (centre for the conditioning and storage of radioactive waste), Marcoule. Could you say a few words about it?

A. MARVY (France): Yes, in the context of the research and development work done in France by the CEA with regard to long term storage issues we have begun to construct a demonstration facility to simulate the design of an underground storage facility for high level waste — that is to say, heat-generating waste. That facility has been designed for tests with regard to the thermal hydraulics of such a facility in order to be able to better simulate the heat generating conditions and determine the necessary requirements in terms of heat removal systems.

K. SHIRAI (Japan): In Japan, some research and development work is being done on tunnel storage because we have many mountains in Japan and it may be of benefit to use the natural environment. Especially, there would be an advantage for protection against seismicity and also terrorist threats.

Summary Panel 1

DISCUSSIONS WITH CHAIRPERSONS OF SESSIONS 1, 2 AND 3

The following discussion was preceded by short presentations by each of the session Chairpersons. The presentations were preliminary versions of the summaries given in the final session of the conference (Session 9). They allowed the conference participants to make comments on or ask questions about the Chairpersons' views of their sessions.

Chairperson: **J. Bouchard** (France)

Members: **A. González** (Argentina) — Chairperson of Session 1
J. Whang (Republic of Korea) — Chairperson of Session 2
W. Brach (United States of America) — Chairperson of Session 3

DISCUSSIONS WITH CHAIRPERSONS OF SESSIONS 1, 2 AND 3

W. BRACH (United States of America): I should like to make a comment about the role of the IAEA Safety Standards. I appreciated Mr. González's comments regarding the Transport Regulations. I am a member of the IAEA's Transport Safety Standards Committee (TRANSSC), and I have frequently discussed with US colleagues who are members of the Commission on Safety Standards, the Nuclear Safety Standards Committee (NUSSC), the Radiation Safety Standards Committee (RASSC) and the Waste Safety Standards Committee (WASSC), the role of those standards. The international role of the Transport Regulations is unique, as Mr. González noted. To support the international transport of radioactive material worldwide there must be — and there is — an institution, through the IAEA and the Transport Regulations, that provides a mechanism to ensure the safe (and I would also say the efficient) transport of radioactive material. In some countries the Transport Regulations are adopted almost in their entirety; in other countries (for example the USA) they are used as a basis for the national transport regulations — at both the US Department of Transportation and the US Nuclear Regulatory Commission.

However, there is, I believe, a difference in the role of the standards that relate to nuclear safety, waste safety and radiation safety. In his presentation on safety standards in Session 1, Mr. Lacoste identified the outcome of the Joint Convention discussions that took place just a month ago with regard to the role of the IAEA Safety Standards. He showed an overhead that, in my opinion, correctly summarized the outcome of those discussions — that the IAEA Safety Standards provide a most useful reference and guide and implementing tool to support the assurance of safety — nuclear safety, radiation safety and waste safety. During the Joint Convention review meeting, apparently a number of hours were spent on that particular topic and I would have a reservation if this conference were to take a position on a topic that at the Joint Convention review meeting was the subject of much discussion among the 41 parties to the Joint Convention. But I endorse Mr. González's comment that the IAEA Safety Standards provide a very useful tool — they provide a means of ensuring that in the nuclear, radiation and waste safety areas there are programmes for providing for the appropriate protection and safety of regulated activities.

A. GONZÁLEZ (Argentina): Before making technical comments, perhaps it is important to comment on your comments. I think that we are all here in our personal capacities, not representing our governments — we are here as scientists, technical people and professionals to discuss technical issues and the IAEA is seeking our professional advice. If we were representing our governments we would be sitting behind our respective country name plates. At the Joint Convention review meetings it is very different; the participants do represent their governments. Here we are representing ourselves with our professional integrity and our beliefs as to what is right, and I consider it important that we all understand that.

Moreover, our comments — good or bad — will be reflected in findings which will be transmitted to the IAEA.

Now to the technical details. As I said, I believe that radioactive material transport provides a good example of what the international community can do to harmonize safety standards. Maybe that example is applicable in the spent fuel management area, maybe it is not, maybe it is partially applicable — this is something that we should advise the IAEA about. If it is applicable, many things will have to happen. First of all, it will be necessary to produce additional safety standards for spent fuel management (at the moment there are very few). If it is applicable, the standards will have to be detailed — and let me say that not just the Transport Regulations fit that description; the Radiation Safety Standards (the BSS) are equally detailed — and are equally followed around the world. There is only one country that does not follow the IAEA Radiation Safety Standards, and that is the USA. The radiation safety standards of the European Union, for example, are — mutatis mutandis — more or less the same as the BSS, and pursuant to the Treaty of Rome they are mandatory in the countries of the European Union, containing precise numbers that must be complied with.

Do we want such detailed safety standards in the spent fuel management area? We need to tell the IAEA. In my opinion it would be very difficult to have real partnerships in that area without such a clear safety framework.

C. GOETZ (Germany): My views regarding the Joint Convention are closer to those of Mr. Brach than to those of Mr. González. We have decided to use the IAEA Safety Standards in the preparation of reports for the next Joint Convention review meeting more than in the past, and I think the decision represented a good and necessary compromise.

In this connection, I would mention the WENRA (Western European Nuclear Regulators' Association) process that started a few years ago and will lead, by 2010, to a harmonization among the WENRA members. In my view this is a very important process because it covers nuclear safety and also radioactive waste management, including spent fuel management.

Within the European Union, we are discussing the idea of a more harmonized regulatory system. The European Commission's Working Party on Nuclear Safety also considers questions of radioactive waste management.

So, from my point of view the IAEA standards are not only background material for the drafting of national regulations but also material for a more harmonized international regulatory approach.

S. JAIN (India): Mr. González has indicated that safety standards should be backed by appraisals, which means international inspections. Are there not too many areas to be covered by an organization like the IAEA — radiation safety, waste safety and so on? With regard to transport, when shipments cross national borders there are inspections, but when they take place entirely within a particular country it is left to the local regulatory bodies to ensure compliance with the national safety standards, which are in line with the international standards.

A. GONZÁLEZ (Argentina): Regarding the comments of Mr. Goetz, at the latest Joint Convention review meeting I represented my government. If I were representing my government here I would agree completely with Mr. Brach. I would say that we arrived at an excellent compromise and we should respect that compromise and that I am not here to undermine it. But my government cannot tell me what to think, and I think that with the Joint Convention we have failed to utilize the mechanism to the full. If we want a Joint Convention, let us have a Joint Convention and be serious about the standards which we apply rather than meeting every two years for superficial discussion and compliments.

As regards appraisals, I did not say that the IAEA 'shall' appraise States' activities. I said that appraisals could be offered as an IAEA service. In the transport area, good use has been made of such a service (called TranSAS) by Brazil, France, Japan, Panama, Turkey and the United Kingdom, and these countries have been very pleased with the results as the TranSAS missions helped them in addressing political problems in the transport area and generated a great deal of useful information.

So it is not that the IAEA 'shall' carry out appraisals. At the request of States, it can do so, and it is for States to decide whether they stand to benefit from an IAEA appraisal.

J. BOUCHARD (France — President): As Mr. González said, we are here as scientists, but we must also be pragmatic. In Session 1, Mr. Lacoste said that most countries are not ready for an effective, comprehensive system.

However, we need to make progress, step by step. Mr. González indicated that the situation is satisfactory in the transport area; in the radiation safety area the situation is satisfactory also. And with time the situation may become satisfactory with regard to reactors, fuel cycle facilities and spent fuel management.

In addition, we need to be sure that we can deal with the expected new nuclear energy technologies. From the presentations made yesterday by Mr.

Brown of the USA, Mr. Zrodnikov of the Russian Federation and Ms. Fouquet of France it is clear that we are heading for new fuel cycle types, new reactor types and new ways of managing spent fuel, so must take steps to develop the necessary safety standards.

J.-C. NEUBER (Germany): We heard something about criticality safety. Our Chairperson (Mr. Whang) said that there is not enough validation of the tools to determine sensitivity and uncertainty. That is not true; there is a lot of validation being done, in particular because it is very simple thanks to the fact that there is a sensitivity tool — you can take two systems which you know are equal or very similar and you can compare the answers which the systems give you — are they similar or not? If the answer is 'no', then the programme is wrong — otherwise the programme is OK.

J. WHANG (Republic of Korea): I really meant to say that validation and verification are not enough or that there is no experiment to provide a validation. But for some cases, like some Eastern reactors such as RBMKs, we have no means of testing results and we cannot be sure whether applying US computer codes to WWERs is appropriate. I would like to hear your opinion.

J.-C. NEUBER (Germany): Russian experts have applied burnup credit to RBMK fuel, so they have some data, which are unknown to us, but that is more a problem of information sharing. I think it would be useful if the IAEA were to inform the community what is available.

J. NISAR (Pakistan): Mr. González rightly pointed out that we should think in a more professional manner instead of representing our countries' interests. I think this principle should be extended to the flow of information from developed to developing countries.

A. GONZÁLEZ (Argentina): I agree completely. An essential element of any safety regime is provision for the application of the safety standards, and that means fostering information exchange, in addition to the transfer of the necessary technology, through technical cooperation. Without it you will not achieve a solid partnership.

E. KUDRYAVTSEV (Russian Federation): I have been asking myself what the limiting factor is for the lifetime of a spent fuel storage facility? For example, is it the fuel integrity? Is it the lifetime of the barriers that prevent the dispersion of material and activity to the environment? Or should we consider the entire system, including the fuel, the barriers, the personnel and so on?

W. BRACH (United States of America): That is a very important question. We have talked about technical considerations — for example, the suitability of the spent fuel for long term dry cask storage, continuing maintenance of the spent fuel while it is in the canisters, and the different cladding materials. Besides technical considerations there are institutional ones — for example, the ability of the organization responsible for the facility to provide the appropriate security, quality assurance, ageing management and so on.

A. GONZÁLEZ (Argentina): I agree that it is a very important question, but I do not think that we have an answer because each one of us has a different answer depending on personal perceptions. I talk about the need for a strong international safety regime given the possible establishment of international partnerships because the only way to answer Mr. Kudryavtsev's question is to arrive at an international agreement on how to answer it. We can have different perceptions but in the end, if we really want an international partnership we need a common answer to his question.

It is like the question about carbon-14. Should we control carbon-14 or not? Your answer will depend on your personal perception. If you have a strong perception of your responsibility towards the future and you are concerned about all the radiological commitments to the future, you have to control carbon-14. If you do not, you release carbon-14.

The truth may lie somewhere in the middle. I do not know what will happen in the end, but an international agreement is essential for answering this kind of question. Without it there will be no international partnership and everyone will continue doing whatever they want to do.

J. BOUCHARD (France — President): When we have a technical question we can easily find an international answer. With a policy question, and Mr. Kudryavtsev's question was partly a policy question, we are not yet ready to provide a completely international answer.

STORAGE FACILITIES

(Session 4)

Chairperson

T. SAEGUSA
Japan

CONCEPTUAL DESIGN OF A MODULAR SYSTEM FOR THE INTERIM DRY STORAGE OF PHWR ATUCHA SPENT FUEL

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Abstract

The reduced storage capacity available in the two spent fuel pools of the PHWR Atucha-1 power plant and the current plans for extending the reactor's operation beyond its design lifetime have, amongst other things, motivated the evaluation of a dry storage option for the interim management of spent fuel assemblies. Two different design concepts are presently being analysed by an expert working group, from both the technical and economic points of view. The authors are proposing a modular system consisting of an arrangement of reinforced concrete structures into which welded metallic canisters loaded with spent fuel assemblies are stored in a horizontal position. The paper describes the modular system and its advantages, and shows how it can comply with the necessary safety requirements.

1. INTRODUCTION AND BACKGROUND

The Atucha-1 nuclear power plant has been generating electricity since 1974 and there is a plan to extend the reactor's operation beyond its 40 year design lifetime. Atucha-1 spent fuel assemblies (SFAs) are presently stored under water in two fuel pools at the reactor site, where more than 9000 SFAs are already being stored on the racks, most of them corresponding to the original fuel design with natural uranium (NU), and the rest containing slightly enriched uranium (SEU), i.e. 0.85 wt.% U-235. Reactor internals such as control rods and fuel channels that were removed from the reactor core for different reasons are also stored in the pools.

The wet storage capacity of the Atucha-1 plant was recently expanded by increasing the SFA density in the pool racks, i.e. re-racking procedures, in order to permit the plant life extension beyond 2014. However, it is expected that the storage capacity of the two fuel pools will be exhausted by the end of 2016. For this reason, an expert working group has been created to evaluate an out of pool dry storage option for the interim management of Atucha-1 SFAs. The working group includes personnel from the utility owner, a state company called Nucleoeléctrica Argentina (NA-SA) that is responsible for the spent fuel management during the reactor lifetime, and from the national Atomic Energy National Commission (CNEA) that will manage the storage of Atucha-1 spent fuel after reactor closure, according to current legislation.

The working group is carrying out a technical and economic feasibility study to evaluate and compare two different design concepts for the Atucha-1 dry storage system. The first concept is being proposed by NA-SA and it is based on the dry storage system implemented by Ontario Power Generation (formerly Ontario Hydro) for its CANDU type spent fuels. In this concept, the SFAs are packaged in casks or containers made from reinforced high density concrete with an external cladding of carbon steel. The same casks are used for fuel loading and on-site transport, and also for long term storage in the storage facility, where they are placed in vertical or horizontal positions. In this concept, the fuel is loaded vertically into the casks in the reactor pools and the casks are then transferred directly from the fuel loading area to the storage site.

The second design alternative for the Atucha-1 dry storage system is being proposed by CNEA and it is adapted from the so-called NUHOMS System (NUteck horizontal modular storage) that is presently used for both PWR and BWR type spent fuels. In this work, the conceptual design of the proposed modular system is presented and the current design and safety analysis activities are described, including thermal and structural calculations of the main safety related components of the system, and criticality and shielding analyses.

2. DESIGN REQUIREMENTS AND CRITERIA

In November 2003 an international workshop was held in Argentina with the main objective of defining the major requirements and criteria to be followed in the conceptual design of the Atucha-1 dry storage system [1]. In the summary report of the workshop [2] the experts attending identified the following main requirements for the dry storage system design:

(a) The spent fuel storage facility should be independent of the reactor, from both the licensing and operational points of view;
(b) The use of existing plant features and equipment should be maximized;
(c) The integrity of SFAS containment should be ensured for at least 50 years;
(d) Fuel conditioning, i.e. draining, drying, helium backfilling, sealing, and leak test in the reactor pool area should be achieved with minimum spent fuel movements and secondary waste generation;
(e) Spent fuel package retrieval from the storage system should be feasible;
(f) Passive cooling systems for the removal and dissipation of the fuel decay heat during storage should be provided, with the following maximum temperature limits at normal operating conditions: (1) 200ºC for fuel rod ZRY-4 cladding, (2) 100ºC for concrete;
(g) The amount of materials and components to be imported should be minimized;
(h) A full complement of national, regional and international regulations should be considered;
(i) There should be differential treatment for failed spent fuels.

Further discussions with Atucha-1 reactor operators led to additional design requirements and wishes, as follows:

(i) The design should be modular, since a phased construction of the dry storage facility allows the distribution of the cost over the time span when storage is actually required;
(ii) The system should be applicable to the management of Atucha-2 plant SFAs;
(iii) The system components to be immersed in the reactor pools should be entirely metallic;
(iv) The physical cross-section of the in-pool system components should be compatible with the dimensions of the compartment available for fuel loading operations in the reactor pools;
(v) The total weight of fuel loaded containers (including water in the container cavity) should not exceed the maximum lifting capability of the handling crane existing at the reactor pool site, that is 800 kN.

Based on all of these design constraints, with special emphasis on the need to use the existing plant features and equipment to minimize the required investment, the authors have concluded that the optimum dry storage design alternative for the Atucha-1 power plant is the modular system that is described in the following.

3. CONCEPTUAL DESIGN OF THE MODULAR SYSTEM

The dry storage system for the Atucha-1 plant must be designed to provide safe confinement for the SFAs, shielding, criticality control and passive heat removal independent of any other facility structures or components. For this purpose, the proposed modular system consists of four major components:

(1) A metallic canister with an internal basket assembly which holds 37 intact SFAs arranged in a triangular packed configuration;
(2) A reinforced concrete storage module in which a single loaded canister is stored in a horizontal position resting on a railed support structure;
(3) A metallic transfer cask that provides physical protection and shielding for the canisters during transfer operations from the plant fuel/reactor building where the canisters are loaded with the SFAs to the independent storage facility, where they are stored in the storage modules;
(4) A heavy vehicle for the on-site transport of the transfer cask containing the loaded canister, that is provided with a trailer for supporting the transfer cask in a horizontal position and aligning it with the storage module, and a hydraulic ram for pushing the loaded canister into the module. This transfer system should interface with the existing plant fuel pools, the handling crane, the site infrastructure, i.e. roadways and topography, and other site specific conditions and procedural requirements.

Auxiliary equipment such as a vacuum drying system and an automatic welding device are also required for the canister loading, draining, drying, helium backfilling and sealing operations. In this concept, the SFAs are loaded into the canister that is placed inside the transfer cask in the fuel pool at the reactor site. The transfer cask containing the loaded canister is then removed from the fuel pool and placed in the decontamination area where draining, drying and sealing operations are performed. The canister cavity is finally backfilled with helium and a leak test is carried out. The transfer cask is then placed on the transport vehicle and moved to the storage facility located on-site. After proper alignment with the storage module the loaded canister is pushed out of the transfer cask into the module using the hydraulic ram. Once inside the storage module the canister is passively cooled by air circulating in natural convection. The horizontal transfer of the loaded canisters into the storage modules obviates the need for a critical heavy lift at the storage location and results in a passive, low profile, impact resistant storage structure. Furthermore, it provides an easy means for canister retrieval and eventual off-site shipment in a licensed shipping cask.

The metallic canister is a high integrity stainless steel vessel for the long term confinement of 37 intact Atucha-1 SFAs, which are located in three concentric rings of 6, 12 and 18 fuel assemblies, respectively, around a central assembly position, as shown in Fig. 1. This triangular arrangement was selected for compactness and suitable self shielding properties, and in order to satisfy the previously mentioned design constraints of cross-section area and weight of in-pool system components.

The canister body consists of a full penetration welded cylindrical shell with an integrally welded, stainless steel bottom closure assembly, and a stainless steel top closure assembly that includes a vent and drainage system. The bottom and top closure assemblies consist of inner and outer cover plates welded to the canister's cylindrical shell and two thick shielding end plugs made of carbon steel, which provide the top and bottom axial shielding required for reducing the occupational doses during canister drying and sealing operations and transfer of the fuel loaded canister to the storage module, respectively. After draining and vacuum drying operations the canister is pressurized above atmospheric pressure with helium gas in order to avoid air in-leakage, to improve the heat transfer and to prevent the oxidation of fuel cladding and, eventually, fuel pellets. The double seal welded containment boundary ensures

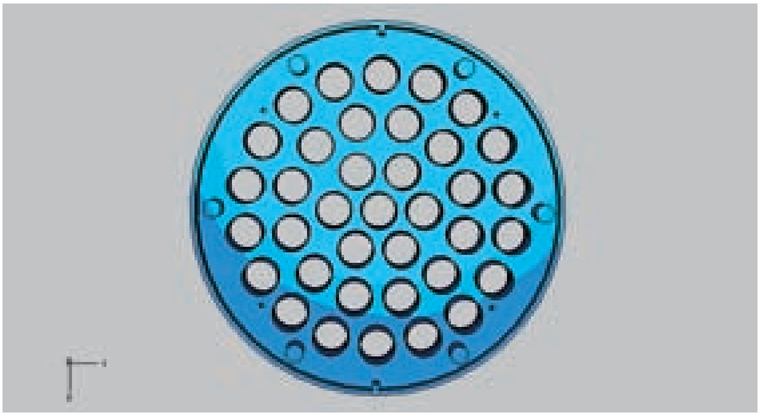

FIG. 1. Cross-section of an Atucha-1 canister showing the triangular array of 37 SFAs.

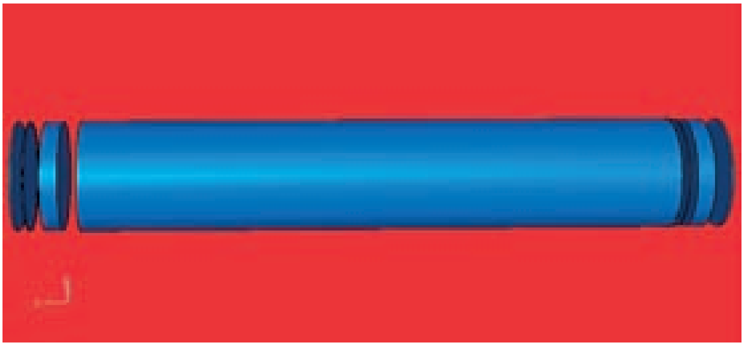

FIG. 2. Components of the Atucha-1 canister body assembly.

that the helium atmosphere is maintained during storage. Figure 2 shows the components of the canister body assembly.

The canister also contains a non-pressure retaining internal basket assembly for providing criticality control and the radial structural support of SFAs, which is shown in Fig. 3. The basket assembly is formed by 37 SFA guide tubes that are held in position by ten axially distributed spacer discs. The axial position of the spacer discs is maintained by preloaded spacer sleeves located around the periphery of each disc. The spacer sleeves are preloaded by six continuous support rods which span the full length of the basket assembly within the spacer sleeves and discs. The fuel guide tubes are mechanically fixed to the lowest spacer disc and are free to move axially with respect to the remaining nine spacer discs. They are open at each end and, therefore, longitudinal fuel assembly loads are applied directly to the canister body and not on the fuel basket structure.

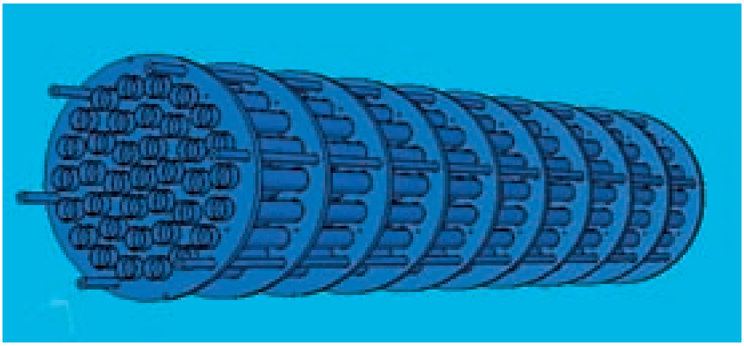

FIG. 3. Perspective of the internal basket assembly.

The reinforced concrete storage module is a massive low profile impact resistant structure designed to withstand all normal condition loads as well as abnormal condition loadings postulated to occur during the design basis accident. It also provides the necessary biological shielding and means to remove the spent fuel decay heat by a combination of radiation conduction and natural air convection. Figure 4 shows schematically the loaded canister located inside the storage module in a horizontal position and resting on a railed structure manufactured from structural steel. The ambient air enters the module through ventilation inlet windows located in the lower side walls, circulates around the loaded canister and exits the module through outlet openings in the upper side walls. Multiple storage modules can be grouped together to form arrays that provide the needed storage capacity consistent with available site space and reactor fuel discharge rates.

On the other hand, the on-site transfer cask is a robust and heavy overpack in which the canister is housed before loading the SFAs. It is designed to provide radiation shielding and structural protection from potential hazards during the canister closure operations and transfer to the storage module. The transfer cask is manufactured from two concentric cylindrical steel shells with a bolted top cover plate and a welded bottom plate. The annulus formed by these two steel shells is filled with cast lead and the outer structural steel shell has a highly polished surface finishing to facilitate decontamination. Four support trunnions are welded to the outer structural shell for pivoting the transfer cask from/to vertical and horizontal positions on the transport trailer.

Finally, the on-site transport system consists of a heavy industrial trailer dedicated to the transport of loaded transfer casks between the plant's fuel/reactor building and the modular storage facility. The trailer is designed to ride

FIG. 4. Schematic view of the loaded canister stored horizontally inside the reinforced concrete storage module.

as low to the ground as possible to minimize the transfer cask height during transport and transfer operations and it is equipped with a hydraulic ram for pushing the loaded canister into the storage module, as well as for the retrieval of the canister if necessary. A specially designed grapple device located at the bottom of the canister is used for coupling with the hydraulic ram.

4. CURRENT DESIGN AND SAFETY ANALYSIS ACTIVITIES

4.1. Thermal and structural design of metallic canisters

Table 1 shows the main physical and nuclear characteristics of PHWR Atucha-1 spent fuels in comparison with two typical LWR type fuels for which the NUHOMS System has been designed, the B&W 15 × 15 PWR fuel, and the GE 7 × 7 BWR fuel.

TABLE 1. MAIN CHARACTERISTICS OF ATUCHA-1 SPENT FUEL TO BE STORED IN THE MODULAR SYSTEM

Parameter	PWR fuel	BWR fuel	PHWR fuel
	B&W 15 × 15	GE 7 × 7	Atucha-1
Physical parameters			
Assembly length (unirrad) (mm)	4210	4467	6029
Assembly width (unirrad) (mm)	216.8	138.1	107.8
Assembly weight (kN)	6.90	3.07	2.07
Heavy metal weight (kg-U)	475.0	194.9	156.3
Fuel cladding	Zry-4	Zry-2	Zry-4
Nuclear parameters			
Fuel initial enrichment (wt.%U-235)	≤4.0	≤4.0	0.71 NU 0.85 SEU 0.71≈ 0.85
Fuel burnup (MWd/MTU)	≤40 000	≤40 000	≤6000 NU ≤12,500 SEU
Cooling time (years)	≥5.0	≥5.0	≥10.0
Decay heat (kW per assembly)	≤1.0	≤0.37	≤0.06

It can be appreciated that Atucha-1 fuel assemblies are structurally much more slender than LWR fuel assemblies and this fuel characteristic has been considered in the canister design in different ways. Firstly, adequate initial clearances were provided between SFAs and guide tubes, and between the basket assembly and the canister cavity, to allow the free thermal expansion of components during system operation. Secondly, the number and axial positioning of spacer discs were selected to avoid excessive stresses and lateral deflections of SFAs during transfer operations and long term storage. Finally, special attention was given to the safety margins against the buckling phenomenon of the canister cylindrical shell.

The canister is designed by analysis to meet the stress intensity allowables of the ASME Boiler and Pressure Vessel Code [3], Section III, Division I, Subsection NB for the canister pressure boundary components, and Subsection NF for the internal basket assembly components. In addition to stress criteria, buckling of the canister shell is evaluated using the ASME Code Subsection NB for Service Levels A, B and C, and ASME Code Appendix F for Service Level D stability criteria.

For the canister stress analysis the load combinations include dead weight loads, thermal loads, internal pressure loads, and handling and lifting loads with the canister placed inside the transfer cask (both in vertical and horizontal positions), and inside the storage module. In particular, the nominal thickness of the canister shell was calculated for a maximum hypothetical internal pressure arising from the 100% failure of fuel rods and assuming 100% release of helium filling gas and 30% release of fission gases and volatiles, at a canister cavity temperature of 200ºC (maximum allowable long term temperature of fuel rod cladding).

From the thermal analysis point of view, Table 1 shows that the maximum heat load expected for Atucha-1 canisters loaded with 37 SFAs is only 2.22 kW, and is mainly due to the low average burnup of PHWR fuels and the long cooling time in the reactor pools. This heat load value is much lower than the expected heat load for LWR canisters, which are typically designed to reject the decay heat arising from 24 PWR type fuels (24 kW per canister) or 52 BWR type fuels (19.24 kW per canister). Nevertheless, a thermal evaluation is required because the temperature limit for Atucha-1 fuel rod cladding is significantly more restrictive than the temperature limits accepted for LWR type SFAs, i.e. 200ºC, compared to 340–350ºC under normal operating conditions.

Thermal analyses have been performed in two different steps. Firstly, a numerical simulation of the passive cooling conditions inside the storage module is carried out using computational fluid dynamics (CFD) methods. A numerical model was developed that consists of three primary components: the spent fuel loaded canister, the concrete storage module, and air duct pathways.

Air is treated as an ideal gas and the 2.22 kW heat load from the 37 SFAs is applied as a constant heat flux on the cylindrical canister surface and over the length of the fuel cavity. In this simplified model the heat exchanges within the canister, including canister shell conductivity, are not represented. It is expected that the CFD simulation will provide a significant insight into local flow patterns inside the storage unit and help to verify that the surface temperature limit allowed for reinforced concrete material, i.e. 100ºC, will be respected.

In addition, the evaluation of temperature distribution inside the fuel loaded canister has been made using the finite element ABAQUS computer code [4]. A three dimensional model of a half symmetry (180º) canister segment was built up for thermal analyses and is shown in Fig. 5. In the model, the spacer discs are represented as half the actual thickness and symmetry boundary conditions in the axial direction are applied on their mid thickness. Air temperatures around the canister are calculated from the previous step are applied as boundary conditions on the cylindrical canister outer surface.

The three dimensional model simulates the SFAs as homogenized solid material occupying the volume within the basket and with an effective thermal conductivity determined on the basis of existing experimental data. Within the canister model, heat is assumed to be transferred by conduction through fuel regions and by gaseous conduction through the gaps between fuels and guide tubes. The heat is then assumed to be transferred to the canister inner surface by conduction through the basket assembly structure and by conduction and radiation through the canister free volume occupied by helium gas (natural convection is neglected) and, finally through the radial gap between internal basket and canister.

Preliminary results of thermal calculations with this finite element numerical model have shown that the design requirements regarding

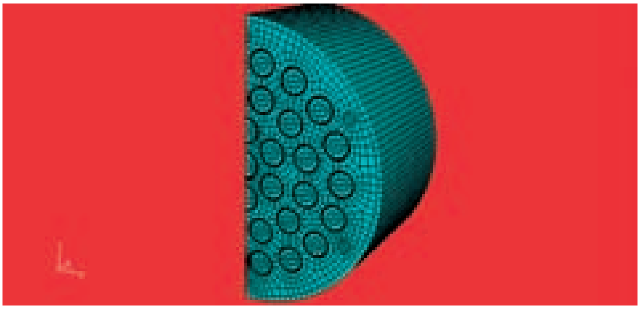

FIG. 5. Finite element mesh of half symmetry canister segment for thermal analyses.

temperature limits under normal operating conditions are fulfilled. A further analysis of temperature distribution in the case of an accident situation, where the storage module cooling windows are fully blocked, is also in progress.

4.2. Criticality and shielding analyses

The canister design criterion for criticality is that an upper subcritical limit of 0.95 minus benchmarking bias and modelling bias shall be maintained for all postulated arrangements of spent fuel within the canister ($k_{eff} \leq 0.95$), in accordance with the regulations of the US Nuclear Regulatory Commission [5, 6].

Criticality analyses were carried out with the MCNP computer code [4] for the conservative case in which the canister is loaded with 37 fresh SFAs (burnup credit is not taken) of either natural uranium or slightly enriched uranium, and two different filling materials in the canister cavity: water (simulating the fuel loading conditions) and helium gas (simulating the long term storage conditions). The results of criticality calculations are presented in Table 2 where it can be observed that, as expected, the highest multiplication factor corresponds to the canister loaded with slightly enriched uranium fuels and filled with water but, in all cases, the calculated k_{eff} is well below the design limit. Additional calculations were performed for hypothetical accident conditions in which the spacer discs lose their structural function completely and the separation among SFAs is then progressively reduced. For the most critical situation, where the 37 fuel guide tubes are in hard contact (minimum theoretical pitch between SFAs), the neutron multiplication factor is 0.5901 ± 0.0005 — that is far away from the design limit.

Radiation shielding evaluations for the proposed modular system were also performed to define the minimum thickness required for the axial top and bottom end plug assemblies of the canister body, the transfer cask and the storage module, in order to reduce to as low as reasonably achievable the

TABLE 2. RESULTS OF CRITICALITY ANALYSES FOR THE ATUCHA-1 CANISTER LOADED WITH NU AND SEU FUELS

Cavity filling material	$k_{eff} \pm \sigma$ Natural uranium	Slightly enriched uranium
Water	0.3975 ± 0.0004	0.4392 ± 0.0004
Helium	0.1286 ± 0.0002	0.1304 ± 0.0002

occupational radiation exposures during canister drying and sealing operations, their transport to the storage facility and the final storage inside the concrete modules.

Shielding calculations were carried out with the MCNP code using source terms estimated with the ORIGEN2 computer code [8] for two different fuel burnups: 6000 MWd/t U (corresponding to the maximum burnup of natural uranium fuels) and 12 500 MWd/t U (corresponding to the maximum burnup of slightly enriched uranium fuels).

Preliminary calculation results show that, for the most reactive array of fuels to be stored in the modular system, a 100 mm thick shell of cast lead is required to reduce the maximum total (gamma plus neutron sources) dose rate on the transfer cask surface to below 1 mSv/h, while a 900 mm thick wall of concrete is additionally needed to reduce the total dose rate below 0.2 mSv/h at one metre from the storage module surface. With these preliminary calculated thickness values, an estimation of the weight of the in-pool system components was performed and the results are presented in Table 3.

From Table 3 the total weight of the loaded canister (wet) located inside the transfer cask is lower than the maximum capability of the reactor pool handling crane (800 kN), and so the design requirement is fulfilled.

5. CONCLUDING REMARKS AND FUTURE WORK

In the framework of the current technical and economic feasibility study to define the best option for the dry storage system to be implemented at the

TABLE 3. SUMMARY OF CALCULATED WEIGHTS OF THE IN-POOL SYSTEM COMPONENTS

Component description	Calculated weight (kN)
Canister body assembly	71
Internal basket assembly	52
Total empty canister weight	123
37 PHWR Atucha-1 fuel assemblies	77
Total loaded canister weight (dry)	200
Water in loaded canister	46
Total loaded canister weight (wet)	246
Transfer cask empty weight	496
Total loaded transfer cask weight	742

Atucha-1 power plant, the authors are proposing a modular system adapted from the LWR fuel NUHOMS system. The modular system seems to be the optimum option for providing safe long term confinement, adequate shielding, criticality control and passive heat removal for Atucha-1 SFAs, while the use of existing plant features and equipment is also maximized with this system.

On the basis of established design requirements and criteria the main features of the modular system have already been conceptually defined. The results of preliminary shielding analyses and thermal and structural finite element calculations have contributed to the definition of the dimensions and geometry of the metallic canister assembly, the on-site transfer cask and the reinforced concrete module. More detailed studies and evaluations of the modular system behaviour are in progress, comprising the analysis of hypothetical accident conditions such as transfer cask drop accidents and storage module block vent accidents. The report of the expert working group will be provided to the government, which will make the final decision on the best option to be implemented for the Atucha-1 power plant.

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SAFETY PHILOSOPHY AT ONTARIO POWER GENERATION USED FUEL DRY STORAGE FACILITIES

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Abstract

Ontario Power Generation's (OPG) used fuel dry storage safety philosophy embodies the defence in depth approach to keep radionuclide emissions and radiation dose rates within the regulatory limits and as low as reasonably achievable. The defence in depth approach is accomplished with multiple barriers between the used fuel and the public during each stage of the used fuel dry storage process. Each barrier independently provides a measure of safety towards preventing the release of radioactive material, as well as providing effective neutron and gamma shielding. Safety assessment and operating experience have shown that these barriers provide a reliable safety margin for the operation of OPG dry storage facilities.

1. INTRODUCTION

Ontario Power Generation (OPG) operates the Pickering Waste Management Facility and Western Waste Management Facility, where OPG has been storing 10 year and older used fuel in dry storage containers (DSCs) since 1996 and 2003, respectively. OPG currently has 5800 tonnes of spent fuel stored in 580 containers. A construction licence for a third facility, the Darlington Waste Management Facility (DWMF), was obtained in August 2004 and is expected to start operations in late 2007.

OPG's dry storage safety philosophy embodies the defence in depth approach to keep radionuclide emissions and radiation dose rates within the regulatory limits and as low as reasonably achievable (ALARA). The defence in depth approach is accomplished by means of multiple barriers between the used fuel and the public during each step in the process of storing the used fuel. Each barrier independently provides a measure of safety towards preventing the release of radioactive material, as well as providing effective neutron and gamma shielding.

The intent of this paper is to describe the safety barriers associated with the fuel itself, the DSCs used to store the fuel, and lastly the dry storage facilities (DSFs) used to store the containers. Additional operational barriers will also be discussed. Finally, the results of the safety assessment for the DWMF are given.

2. USED FUEL DRY STORAGE PROCESS

The irradiated fuel discharged from the nuclear reactors is stored in the irradiated fuel bays (IFBs) under demineralized water. After 10 years in the IFBs the irradiated fuel can be transferred to dry storage. A DSC (see Fig. 1) is submerged in the cask handling bay area and four storage modules, each containing up to 96 used fuel bundles, are loaded into the DSC. While inside the cask handling bay area a clamp is used to secure the container lid to the

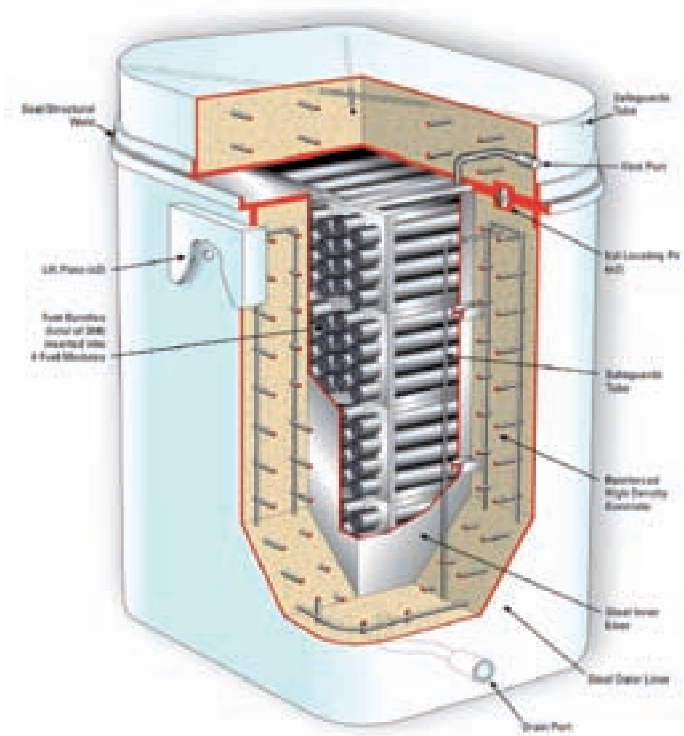

FIG. 1. Dry storage container cutaway figure and stack of four storage modules.

container main body. After the container is lifted from the handling bay it is drained, decontaminated and vacuum dried. The container is then ready to be transferred by means of a specially designed vehicle to the DSF. Each OPG DSF is within a nuclear power plant site.

At the DSF the DSC lid is seal welded to the main body of the container and the weld is inspected using X radiography. After ensuring the integrity of the welds the container cavity is vacuum dried and backfilled with helium. The container is placed in a vacuum chamber and helium leak tested. After this step, permanent safeguards seals are installed. The container is then transferred to the container storage building within the DSF.

3. SAFETY BARRIERS

The design and operation of a DSF is such that under normal and abnormal operating conditions these facilities can be operated safely and without undue risk to workers, members of the public or the environment.

3.1. Used fuel

The fuel bundles used at OPG reactors contain natural uranium CANDU® fuel and are assemblies of 37 or 28 cylindrical fuel elements arranged in concentric rings around a central element (see Fig. 2) and 49.53 cm in length. Each fuel element contains high density natural UO_2 pellets in a zirconium alloy tube sheath.

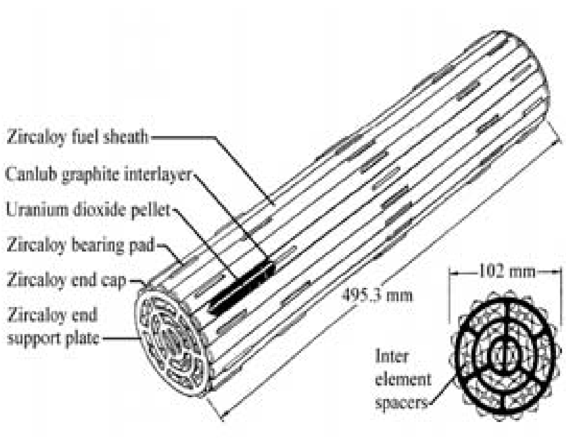

FIG. 2. CANDU fuel bundle.

Only used fuel that has been cooled in an IFB for at least ten years can be considered for dry storage. After this time the inventory of the short lived radionuclides and the decay heat have decreased considerably.

The location of radionuclides in a used fuel element depends on their chemical and physical behaviour and on where they were produced within the fuel element. The majority of new radionuclides, such as the fission products and actinides in ten year cooled used fuel, are embedded within the lattice of uranium and oxygen atoms, very close to where they were produced. They substitute for uranium in the uranium dioxide lattice. Activation products that are produced in the zircaloy sheath are mostly trapped by the zirconium alloy and do not diffuse any significant distance from the site of their formation.

Cracking and grain growth occurs in the ceramic fuel pellets at the high temperatures and temperature gradients in the reactor (400–2000°C). The cross-section of a ceramic fuel pellet is shown in Fig. 3. About 2% of the gaseous radionuclides, or those that are volatile at fuel irradiation temperatures, are released to the cracks in the pellets and to the gap between the pellets and the fuel sheath. A further 6% segregates to the grain (crystal) boundaries within the uranium dioxide pellets. Laboratory studies have been used to predict the amounts of radionuclides in the fuel sheath gap and at the grain boundaries.

The release of radionuclides from failed fuel depends on the volatility of the chemical forms found in the fuel at the maximum fuel temperature and their ability to migrate through the fuel grains. Radionuclides born in the fuel may remain in elemental form or combine with other nuclides, the uranium

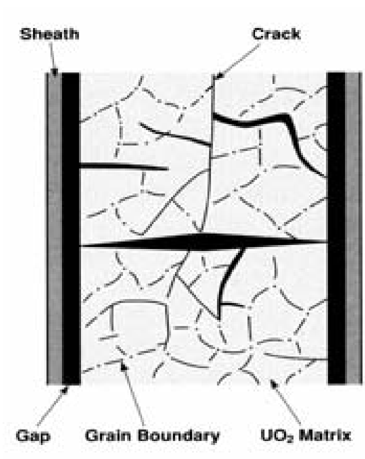

FIG. 3. Locations of radionuclides in used fuel.

dioxide fuel, the zircaloy sheath, or excess oxygen. The chemical forms of fission products have been studied in the past to determine their respective contributions to releases from failed fuel.

The maximum sheath temperature for used fuel with a decay heat of 6.4 W/bundle has been estimated to be less than 150°C in dry storage in a helium atmosphere. At these low temperatures, only the volatile fission products would be released should the fuel sheath become damaged. The volatile fission products include krypton-85 and tritium, mainly in the form of tritiated water vapour (HTO). Release of these radionuclides on sheath failure is taken into consideration when performing a safety assessment.

The uranium dioxide matrix constitutes the first safety barrier since it effectively contains the radionuclides present in the used fuel cooled for 10 years (either under wet or dry storage conditions), except for the tritium (in vapour form) and krypton-85 (which is a gas). The second safety barrier is the fuel cladding or sheath which contains the tritium and krypton-85 that would otherwise be available for release.

Although random visual inspection of fuel coming out of the reactors and more detailed examination of irradiated fuel bundles in hot cell facilities have shown only an average of 0.02% defective fuel, an operational barrier is created by restricting the used fuel transferred to dry storage to fuel which has no visible or known defects affecting the integrity of the fuel bundle, further reducing the possibility of a release.

A further layer of defence is provided by the water in the IFB since the DSC loading takes place inside the bay. The bay water provides neutron and gamma shielding and also traps any tritium that might be released from the fuel cladding in the event of fuel failure.

3.2. Dry storage container

The DSC is a free standing reinforced heavy concrete container with an inner steel liner and an outer steel shell for the storage of used CANDU® fuel. It is made of two sub-assemblies, a lid and a base (see Fig. 1). The base provides the storage space for 384 used CANDU fuel bundles in four storage modules. The container is rectangular in shape with outside dimensions of 2.121 × 2.419 m by 3.557 m in height (including the lid) and an inside cavity of 1.046 × 1.322 m by 2.520 m in height. The thickness of the carbon steel shells is 13 mm and the container walls are made of 52 cm reinforced heavy concrete with a density of 3.5 to 3.7 mg/m^3. The approximate weight of an empty container, including the lid, is 60 mg and approximately 70 mg when fully loaded.

The DSCs are designed to prevent or reduce to an acceptably low level the escape of fission products in the unlikely event that a fuel failure occurs under normal or accident conditions and to protect the fuel from extensive damage from missiles or falling objects during abnormal or accident conditions. Although the DSC was designed to hold 6 year old used fuel, an operational safety barrier is applied by only loading 10 year or older fuel.

The DSC containment system is the first safety barrier provided by the container. It is defined as the inner liner, the bottom plate of the storage lid, the lid locating pin housings (lid and base), the structural/seal weld between the lid and base, and the vent and drain ports. The lid seal weld is a 31.75 mm thick full penetration weld between the 32 mm thick base plate of the container lid and the perimeter flange of the container body. The weld is designed to secure the storage lid in place to provide the container with the required structural strength and to complete the containment barrier for the used fuel. The vent and drain housings have steel shielding plugs that are also seal welded. All welds that form this containment system and all welds attaching items to the containment system are classified as 'nuclear welds'. The welds are inspected according to ASME code specifications and a further helium leak test is carried out before storage.

A second safety barrier is provided by the 52 cm reinforced heavy concrete walls of the container, which effectively provide gamma and neutron shielding as well as structural integrity to prevent any fuel damage during normal dry storage operations or during abnormal or accident conditions.

The 13 mm outer steel shell is another safety barrier. In addition to providing shielding and structural strength it is coated for corrosion protection — to facilitate decontamination of the container following wet loading operations in the fuel bay. All welds on the outer shell are classified as 'structural welds'. An additional safety barrier to preserve the long term fuel integrity is provided by backfilling the DSC with helium prior to storage.

The DSC has been designed to withstand an impact load of 45 608 kN, which is equivalent to ~65 g deceleration. This has been confirmed by quarter scale model drop tests in which the model survived intact under a 250–300 g deceleration, equivalent to ~62–75 g in full scale. In the half scale tests the model survived 230–310 g deceleration, equivalent to ~115–155 g deceleration in full scale.

Although it has been estimated that, if dropped from 2.4 m on to a concrete floor, the deceleration would be equivalent to 40 g, well below the design basis of 65 g, an operational safety barrier is provided by limiting the normal lift height for a loaded unwelded container with transfer clamp installed to about 20 cm and for a seal welded container to a maximum of about 1.3 m.

A specially designed clamp to securely attach the lid to the container base is used during the transfer of a loaded container from the fuel bay to the DSF. The clamp is used in conjunction with an elastomeric seal between the lid and the base to permit the cavity of the loaded container to be vacuum dried and left under slightly negative atmospheric pressure during the on-site transfer.

The transfer clamp has been designed to prevent the lid from separating in the event that the DSC rolls over. The transfer clamp has also been designed to maintain its integrity in the event of tornado winds and tornado generated missiles.

The transfer clamp provides a safety barrier during transfer, minimizing the chances of releases to the environment and keeping the fuel bundles contained within the container cavity in the case of a credible postulated abnormal or accident event.

3.3. Dry storage container transfer vehicle

The vehicle used to transfer the DSCs from the fuel bays to the DSF is a specially designed multi-wheeled vehicle (see Fig. 4). The vehicle is self loading and self powered by a diesel engine. The vehicle is designed to operate at a pre-determined maximum safe speed. The vehicle's hydraulic system raises it and engages the lift trunnions in the container lift plates, and raises the container from the ground. Once lifted the suspension is mechanically locked, preventing the container from being inadvertently lowered to the ground in the event of hydraulic failure.

The vehicle design has a preventive safety barrier in the form of a limit on the height above ground (20 cm) at which the container is carried. This is to mitigate the consequences of a credible postulated abnormal or accident event.

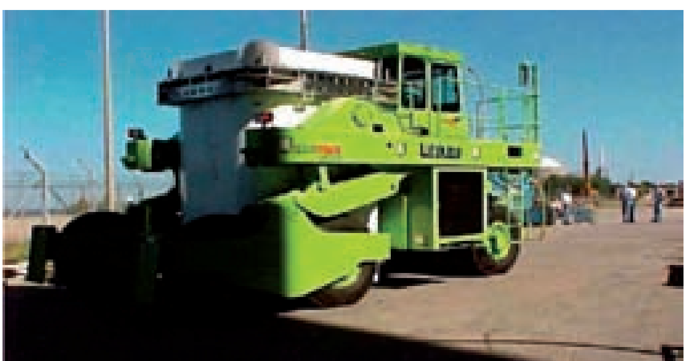

FIG. 4. Dry storage container transfer vehicle.

3.4. Dry storage facility

Each DSF consists of a DSC processing building and up to 4 DSC storage buildings (see Fig. 5). In the container processing building the containers are processed to prepare them for storage. The maximum capacity of each DSC storage building is 500 containers. Their construction is staged as additional storage space is required. The DSC storage building walls are made of ordinary concrete and the buildings are provided with passive ventilation systems that ensure an ambient temperature around the containers not higher than 38ºC.

The thickness of the concrete walls was determined assuming that the facility is full to capacity with fully loaded containers containing the ‘reference used fuel bundle’ with a burnup which is 95% of the maximum burnup of the fuel discharged from the reactors of that particular nuclear power plant. This is a very conservative assumption for the design but provides additional shielding to further reduce the gamma radiation doses to the public. Ontario Power Generation (OPG) also imposes another safety margin on the the facility design, a target dose limit for members of the public at the nuclear power plant site boundary of less than 10 μSv/a, which is 1% of the regulatory limit for members of the public.

An additional operational safety barrier is added by placing the containers loaded with older fuel close to the perimeter walls of the storage building, which reduces further the radiation dose rate to the public at the site boundary.

4. SAFETY ASSESSMENT

During the entire used fuel dry storage operation the focus is on the prevention of fuel damage to ensure that fission products remain contained

FIG. 5. Dry storage container processing and storage buildings.

within the fuel elements. If an abnormal event or accident occurs the used fuel dry storage process provides mitigation and accommodation measures to ensure that sufficient barriers remain intact so the radioactive releases will be below the Canadian Nuclear Safety Commission (CNSC) regulatory limits.

The ultimate evaluation of the safety of a DSF is done by means of a safety assessment. A safety assessment of each facility and the associated systems and operations is required to support each request for regulatory approval to construct and operate a facility. The objective of the safety assessment is to assess the radiation doses to the public and the workers under normal operation and postulated credible accident scenarios and to confirm that they are below the regulatory dose limits for the workers and members of the public.

4.1. Normal operating conditions

For the safety assessment of a particular facility, a reference used fuel bundle is defined based on the operating history and data on fuel discharged from the reactors of the specific nuclear generating plant.

The primary factors that determine the characteristics of the used fuel are the physical attributes, power and burnup histories, and decay time. These factors are in turn influenced by fuelling strategies and reactor conditions. Therefore a reference fuel bundle is defined for the purpose of performing the safety assessment of a given facility and processes. The reference used fuel bundle is defined based on the burnup histograms of used fuel over 5 years, such that it conservatively represents the used fuel bundles discharged from the reactors at the particular nuclear power plant. Thus the safety assessment based on the defined reference fuel bundle generates conservative results. The reference fuel bundle is assumed to have been out of the reactor core for 10 years. The burnup of the reference fuel bundle is chosen conservatively and the burnups of 95% of the fuel bundles discharged from the station are below the burnup of the reference fuel bundle.

Figure 6 shows the radiation dose rates from a single container filled with 384 reference fuel bundles for the DWMF. It shows the dose rate versus distance from the facility wall. The facility was assumed to have 1500 DSCs filled with 10 year old reference fuel bundles. Operating experience at the Pickering and Western waste management facilities has shown that the predicted radiation dose rates are conservative. For containers loaded with 10 year cooled or older fuel, measured contact dose rates to date are about 9 to 13 μSv/h. This compares with estimates of 67 μSv/h contact dose rates for fuel cooled for 10 years (at the container side or front). At a 1 m distance, measured dose rates are about 5 to 7 μSv/h, compared with calculated dose rate estimates

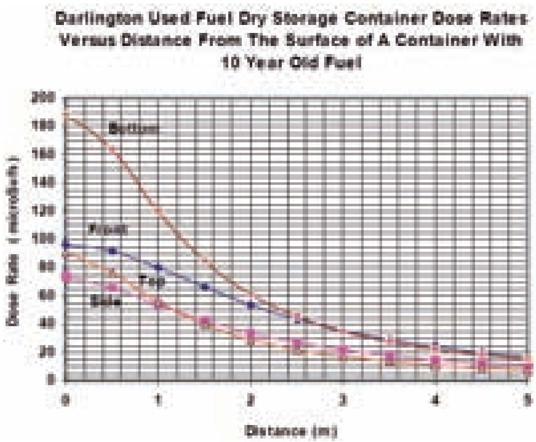

FIG. 6. Radiation dose rate versus distance from the surface of a single dry storage container with 384 reference used fuel bundles cooled for 10 years.

of 48 to 56 μSv/h. The dose rate at the plant site boundary, assuming the facility is full, was calculated to be 3.3×10^{-2} μSv/a, well below OPG dose rate target of 10 μSv/a.

A thermal analysis of a container loaded with fuel bundles with a decay heat of 7.4 W per bundle was carried out. An ambient temperature of 38°C in the proximity of DSCs was assumed. The maximum internal and external container surface temperatures were calculated to be 94°C and 58°C, respectively. The stresses generated in the concrete by this thermal gradient have been assessed to result in no significant cracking of the concrete over a storage period of 50 years. However, only 10 year or older fuel is actually loaded into the containers, with an 8% lower heat load.

4.2. Malfunctions and accident conditions

When assessing malfunctions and accidents, postulated external and internal events are considered. Consideration is also given to the design basis accidents of the specific nuclear generating plant that could affect the used fuel dry storage operations as the waste storage sites are in close proximity of the nuclear operating parts of the stations. The probability of occurrence of each initiating event considered is calculated and for those events deemed credible (i.e. frequency > 10^{-7} events per year), a bounding fuel failure consequence is predicted.

In the event that a 10 year or older used fuel bundle becomes damaged during the dry storage operations, as mentioned previously, the only significant radionuclides available for release are krypton-85 and tritium. For a fuel element damaged under abnormal operating conditions it is conservatively postulated that the free inventory of tritium and krypton-85 in the gap between the fuel matrix and the zircaloy sheath, plus 10% of the inventory in the grain boundary, would be released. The gap fraction is assumed to be 0.0365 for tritium and for krypton-85. The grain boundary fraction is assumed to be 0.1209 for tritium and for krypton-85, conservatively assuming that tritium behaves similarly to krypton. For the safety assessment of a postulated malfunction or accident event leading to airborne emissions, the calculation of tritium and krypton-85 inventories takes into account the activation in the reactor core of small quantities of impurities present in the used fuel. Release of these radionuclides is considered in calculating public and worker doses.

Tables 1 and 2 show the estimated public and occupational radiation doses due to the postulated credible malfunctions and accidents during the DSC processing and storage, respectively.

A bounding accident during loading, transfer and processing of the containers was postulated, assuming the failure of 30% of a container's used fuel content, i.e. 30% of the fuel elements in all 384 fuel bundles, a total of 4262 failed fuel elements. Realistically, fuel sheath failure is not expected to result from an accidental container drop from the low lift height of the transporter and during processing in the processing building. The free tritium and krypton-85 in the damaged fuel elements is assumed to be released into the container cavity. Ignoring the barriers provided by the transfer clamp seal and the sub-atmospheric pressure inside the container cavity, it is assumed that these radionuclides are released promptly to the environment. The radiation doses to the public due to this event were assessed to be 1.5 μSv for an adult and 1.1 μSv for an infant at the Darlington site boundary. The dose to an individual in the immediate proximity of the container is assessed to be 4.5 mSv. The estimated radiation doses to the public from this bounding postulated event were less than 0.2% of the regulatory limit.

5. CONCLUSIONS

The safety assessment of the dry storage facilities and the operating experience achieved at the Pickering and Western waste management facilities have shown that the design and operational safety barriers provide a substantial safety margin for the safe operation of the dry storage facilities, without undue risk to workers, members of the public, or the environment.

TABLE 1. CREDIBLE POSTULATED MALFUNCTIONS OR ACCIDENTS DURING DRY STORAGE CONTAINER PROCESSING

Malfunction or accident	Potential maximum radiation dose to the public (μSv)		Potential maximum occupational radiation dose (mSv)
	Adult	Infant	
Drop of a DSC during handling	1.5	1.2	4.5
Equipment drop onto DSC	<1.5	<1.2	<4.5
DSC collision during craning	<1.5	<1.2	<4.5
Transporter collision with loaded DSC	<1.5	<1.2	<4.5
Equipment collision with loaded DSC during craning	<1.5	<1.2	<4.5
Processing building fire	0	0	0
Inadvertent operation of the X ray machine without the DSC in position	$<1.5 \times 10^{-4}$	1.2×10^{-4}	0.16
Inadvertent operation of the X ray machine with someone inside the radiography room	$<1.8 \times 10^{-5}$	1.4×10^{-5}	0.02
Earthquake	<1.5	<1.2	<4.5
Thunderstorms	0	0	0
Toxic corrosive chemical rail line accident	1.5	1.2	4.5
Tritium removal facility explosion	0	0	0
Hazardous material building explosion	0	0	0

TABLE 2. POSTULATED MALFUNCTIONS OR ACCIDENTS DURING DRY STORAGE CONTAINER STORAGE

Malfunction or accident	Potential maximum radiation dose to the public (μSv)		Potential maximum occupational radiation dose (mSv)
	Adult	Infant	
Seal weld failure during storage	0.21	0.16	0.6
DSC drop during transfer to storage	0	0	0
Transporter collision with a DSC	0	0	0
Storage building fire	0	0	0
Earthquake	0	0	0
Tornado	0	0	0
Thunderstorm	0	0	0
Rail line blast	0	0	0
Toxic corrosive chemical rail line accident	0	0	0
Tritium removal facility explosion	0	0	0
Hazardous material building explosion	0	0	0

THE SPENT FUEL STORAGE SOLUTION IN ARMENIA

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Abstract

The paper describes the successful adoption of the NUHOMS® system for spent fuel storage, which was developed for application for PWR and BWR fuels, to the fuel of the Armenian nuclear power plant, which is of the WWER type. The storage system is described and the issues which emerged in its adoption to the fuel of a different nuclear power plant system and to a different regulatory and social environment are elaborated.

1. THE NUHOMS® TECHNOLOGY

The NUHOMS® system was initially designed by TRANSNUCLEAR, Inc. for the horizontal dry storage of PWR and BWR spent fuel assemblies in the USA. The system provides safe dry storage and long term interim storage for spent fuel assemblies that have been out of the reactor for a sufficient period of time. The spent fuel assemblies are canisterized and stored in concrete horizontal storage modules.

1.1. Description of the NUHOMS® system

The spent fuel assemblies are confined in an inert atmosphere by a canister containment pressure vessel. The canister is protected and shielded by a massive reinforced concrete module. Decay heat is removed from the canister and the concrete module by a passive natural draft convection ventilation system.

The canisterized assemblies are transferred from the spent fuel pool of the nuclear power plant to the concrete storage module located at the independent spent fuel storage installation in a transfer cask. The cask is aligned with the storage module and the canister is inserted into the module by means of a hydraulic ram. The NUHOMS® system is a passive installation that is designed to provide shielding and safe containment of spent fuel for a range of postulated accident conditions and natural phenomena.

1.2. Components of the NUHOMS® system

The main components and equipment of the system are:

(a) Dry shielded canister (DSC): this canister provides containment of all radioactive materials and encapsulates the fuel in a leaktight and inert atmosphere.
(b) Horizontal storage module (HSM): The HSM provides radiation shielding and physical protection for the canister against a wide range of postulated natural hazards.

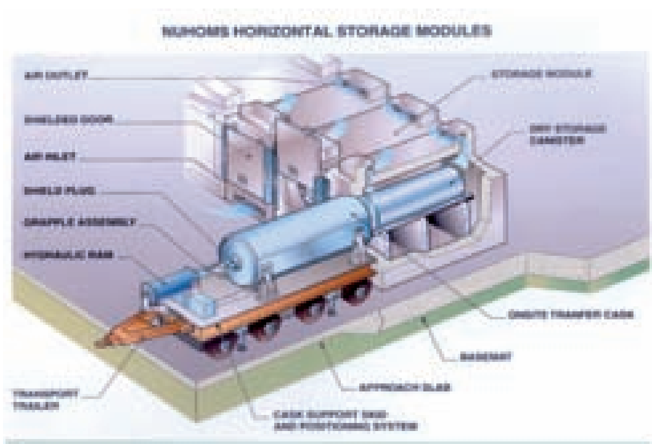

FIG. 1. NUHOMS® horizontal storage modules.

(c) Transfer cask: This provides radiation shielding and structural protection for the DSC during the transfer operation, while providing passive heat removal for the canisterized spent fuel.

1.2.1. Dry shielded canister characteristics

The dry shielded canister consists of a cylindrical shell, top and bottom shield plugs, inner and outer closer plates, and inner and outer top cover plates. The canister provides containment of all radioactive materials, encapsulates the fuel in an inert helium atmosphere (the canister is backfilled with helium before being closed by seal welding).

The cylindrical shell and the top and bottom cover plate assemblies form the pressure retaining containment boundary for the spent fuel. The canister is equipped with two shield plugs so that the intensity of the radiation at the ends is sufficiently reduced to allow safe drying, sealing and handling operations.

The canister has double, redundant seal welds which join the shell and the top and bottom shield plug and cover plate assemblies to form the containment boundary. The bottom end assembly containment boundary welds are made during the fabrication of the canister. The top end assembly containment boundary welds are made after fuel loading. The dry shielded canister is equipped with an internal basket which ensures the safe storage of the spent fuel with regard to the criticality considerations.

1.2.2. Horizontal storage module

The horizontal storage modules are reinforced concrete modules, cast in place or prefabricated. The modules provide the base protection for the dry shielded canister and provide physical protection against all worst case postulated natural phenomena such as tornadoes, earthquakes, and floods. It also provides highly effective biological shielding sufficient to keep occupational radiation exposures and public exposures below regulatory limits.

The horizontal storage modules are expandable concrete structures designed for the long term storage of dry shielded canisters. The module is equipped with hardened stainless steel rails that provide a sliding surface for the canister during loading and unloading operations. The design of the module allows the canister to be transferred and stored without requiring it to be lifted at the independent spent fuel storage installation during initial loading or later when unloading to transport to an operating repository. This is an important economic advantage when considering the life cycle of dry storage.

The module design incorporates passive ventilation for the removal of spent fuel decay heat from the canister. It is ventilated, allowing air to enter the module through the bottom vents, flow around the canister and exit through the top vents. The storage module ventilation system has a heat removal capacity of up to 40.8 kilowatts per canister (the standardized module has been designed for a heat load of 24 kilowatts per canister and the latest, the HSM-H, for a heat load up to 40.8 kilowatts per canister). The NUHOMS® storage module is qualified for external ambient temperatures ranging from 40°C to 51°C. The top vents are provided with vent caps which provide increased shielding.

1.2.3. Transfer cask

The transfer cask provides shielding and protection from potential hazards during the dry shielded canister closure operations and transfer to the horizontal storage module. The transfer cask is designed for on-site transfer of the canister from the fuel pool of the nuclear power plant to the horizontal storage module. The cask is constructed from two concentric cylindrical steel shells with a bolted top cover plate and a welded bottom end assembly. The annulus formed by these two shells is filled with cast lead to provide shielding from gamma radiation.

The transfer cask includes an outer steel jacket which is filled with water or a hydrogen rich solid material, for neutron shielding. The top and bottom end assemblies incorporate a solid neutron shield material. The cask provides sufficient shielding to ensure that dose rates are within regulatory limits.

The shell materials are resistant to corrosion. The transfer cask has been used successfully in PWR and BWR spent fuel pools without exhibiting an adverse interaction with the spent fuel pool water.

The empty canister is inserted into the transfer cask cavity and the cask is then placed vertically in the spent fuel pool. The spent fuel assemblies are loaded into the canister and the transfer cask with the canister is moved from the spent fuel pool to the cask preparation area where the canister is drained, dried and the top cover plates are welded. The loaded canister and the cask are then moved horizontally from the fuel building to the independent spent fuel storage installation on the transfer skid secured on the transfer trailer. The canister is transferred from the transfer cask to the horizontal storage module for storage.

The system has been optimized by standardization of design, fabrication and operation. It has the flexibility to accommodate various canister lengths and therefore various types of fuel. The interchangeability of equipment allows the transfer and auxiliary equipment to be used on multiple sites, allowing users to share operational experience and cut costs.

2. ADAPTATION OF NUHOMS® TECHNOLOGY TO THE ARMENIAN NUCLEAR POWER PLANT SITE REQUIREMENTS

2.1. A solution for Armenia's needs in dry storage

The Armenian nuclear power plant (ANPP) site at Medzamor, 30 km from Yerevan, hosts 2 WWER-440 PWR units. Both units were shut down after an earthquake in 1989. Unit 2 was upgraded and restarted in 1996.

In the past, spent fuel was sent for reprocessing in the former Soviet Union. After the restart of Unit 2, the spent fuel has been continuously accumulated in the spent fuel pools of Unit 1 and Unit 2 because the rail link with the Russian Federation is no longer available.

Due to the unavailability of a back end solution (nuclear fuel reprocessing or a permanent repository) in Armenia, interim storage of spent fuel has become necessary. The spent fuel assemblies were cooled in the ANPP fuel pools. However, as the existing fuel pools began to approach their maximum capacities, new storage capacity was required. In addition, it was recognized that in order to maintain long term spent fuel integrity the fuel must be retrieved from the pools and stored in dry conditions.

In 1996, AREVA NP (formerly Framatome) signed an agreement with the ANPP to supply the NUHOMS® technology for dry storage of the spent fuel accumulated to that date. The implementation of the system at the ANPP site was accomplished in 1998. It included the supply of the overall engineering and licensing details for the dry storage system of the spent fuel assemblies of the ANPP, as well as the supply of canisters and auxiliary equipment (transfer cask, vacuum drying system, leak detector, remote automatic welding system, etc.).

To enable the storage of 616 assemblies on the Medzamor site for a period of 50 years, the construction of 11 horizontal storage modules was decided upon. These modules are grouped to form 2 arrays of 5 and 6 modules,

FIG. 2. Arrangement of horizontal storage modules at the Armenian NPP.

respectively. They are surrounded by a double fence, a security fence and a perimeter fence. The two arrays are located in a back to back arrangement.

The site selected for the construction of the independent spent fuel storage installation is situated within the ANPP site. The size of the site is adequate to accommodate the desired number of modules.

2.2. Adaptations required by site characteristics and other specificities

2.2.1. Canisters adapted to WWER-440 fuel

The standard NUHOMS® system is designed for PWR and BWR fuels and had to be adapted by AREVA to WWER fuels. The canisters were designed to accommodate the current WWER-440 fuel assemblies.

The radiological characteristics of WWER assemblies, with an initial enrichment of 3.6% U-235, an average burnup of 29 000 MWD/Mt U (max. burnup of 42 000 MWD/Mt U) and a post-irradiation cooling time of 5 years, are quite compatible with the standard system technology and so they did not require significant adaptation.

The hexagonal geometry of WWER assemblies required a new design for the internal basket assembly inside the canister. The basket consists of a grid assembly of hexagonal steel guide sleeves that make up 56 fuel compartments. Some guide sleeves are made of boron stainless steel that provides an additional criticality control without relying on moderator exclusion.

2.2.2. Remote automatic welding system

In Armenia, a specific device has been developed for the welding system in order to meet the needs of the ANPP. It is a flexible and easily adaptable device that results in reduced development and maintenance costs. Spare parts for this welding system can be easily found in commercial industry. This automatic system has allowed the reduction of personnel exposures to ionizing radiation by reducing the time required for canister welding operations.

2.2.3. Adaptations required for the construction of the horizontal storage modules

The design and engineering studies carried out for the construction of the modules showed that there were some specific requirements related to the ANPP site characteristics and other specificities:

(a) Consideration of the characteristics of the structural materials such as concrete and steel which have been fabricated locally and used for construction of the modules;
(b) Consideration of the Armenian context and the adoption of civil engineering work for construction of the horizontal storage modules.

In the USA the modules used for dry storage are cast in place or prefabricated off-site. The US power plants tend to use the prefabricated modules because they are easily transported and assembled on the storage sites according to need, and minimize the impact on power plant operations.

In Armenia it was decided to construct the storage modules on the nuclear power plant site for convenience. Since the ANPP is the only nuclear power plant in Armenia it was considered to be more suitable to build storage modules directly on-site since there are no needs for similar devices elsewhere in the country and it was the cheapest solution for the ANPP.

The storage modules have been integrated in a monolithic concrete structure cast on-site.

(1) The seismicity of the Medzamor site is characterized by a horizontal acceleration of 0.4 g. Therefore, the set of 11 modules had to be adapted to the site seismic conditions. Armenian seismic criteria are slightly more stringent than those taken into consideration for the standardized NUHOMS® system.
(2) The dry storage modules were adapted to the length of the canisters containing WWER-440 spent fuel, different from the canisters used for the standardized NUHOMS® system.
(3) For the design of the modules, the thermal characteristics of the WWER spent fuel as well as the climatic data of the Medzamor site were taken into account (heat load per canister 14.8 kW and external ambient temperatures ranging from –40°C to 51°C).

2.2.4. Interface with existing facilities and structures

The NUHOMS® technology is designed to maximize the use of existing plant features and equipment and to minimize the need to add or modify equipment. The ANPP’s independent spent fuel storage installation is located inside the existing plant security boundary so that no separate protected area is required.

In order to limit the costs, the capabilities of existing equipment on the site of the ANPP were studied and the equipment was subsequently used for fuel loading and cask transfer operations. For example, loading of the canisters at the ANPP was accomplished using the existing facilities without modifications to the reactor building; the canister and transfer cask handling inside the reactor building was carried out using the existing crane; the transfer cask transport between the reactor building and the storage yard was carried out using a road trailer and a transfer cask support skid.

3. FEEDBACK FROM THE FIRST OPERATIONS

All the operations, including fuel loading, closure of the dry shielded canister, transfer to the fuel storage facility and canister loading into storage modules were performed safely.

Some lessons were learned concerning four main points:

(1) Safety and security:
   - Radiation dose rates observed during loading and after loading and transfer of the canisters,
   - Events recorded during the same operations;

(2) Time required for canister loading and transfer operations;
(3) Transfer of competences and know-how;
(4) Involvement of local companies.

3.1. Safety and security

3.1.1. Radiation dose rates

The operating procedures for the NUHOMS® system resulted in the minimization of occupational radiation exposures. Radiological protection conformed to the Armenian requirements, which are quite comparable with those of Western countries. The ANPP's existing radiation safety and ALARA (as low as reasonably achievable) policies for the plant were applied to the storage facility. Storage facility personnel were trained and updated on ALARA practices and dose reduction techniques. Implementation of storage facility systems and equipment procedures were reviewed by the ANPP to ensure that exposures during all phases of operations, maintenance and surveillance were ALARA.

The studies concerning the adaptation of the NUHOMS® system to WWER fuels were conducted with the aim of achieving the same level of radiation protection as reached for NUHOMS® applications to PWR and BWR fuels. Follow-up studies have shown that the radiation exposure of site personnel during fuel handling and transfer activities has progressively decreased due to the ANPP teams' increasing competence in the NUHOMS® operating procedures. The latest follow-up on radiation dose rates around the storage site showed that the average dose rate at the storage site fence was less than 0.13 mSv in 2003.

3.1.2. Events during operations

No incident or accidental events occurred during the spent fuel storage operations performed at the ANPP.

3.2. Time required for the NUHOMS® operations

The operation of the NUHOMS® system consists of implementing three basic operational sequences: the loading of spent fuel into the canister, closure of the canister, and transfer of the canister to the storage facility. On the ANPP site, canister loading, closure and its transfer to a storage module took 12 to 15 days.

The loading operations could not be accomplished during December and January because of weather conditions, and during June and July because of shutdown of the reactor. In addition, the Armenian Nuclear Regulatory Authority (ANRA) imposed an additional requirement on the NUHOMS® operations: that they can be performed only if the difference between outside temperature and reactor hall temperature is less than 20°C.

3.3. Transfer of competences and know-how

AREVA was responsible for the storage facility construction. Office and on-site engineering and quality assurance support were provided during manufacturing and erection to ensure that system design, licensing and operational requirements were met.

A complete operational test was performed on-site after delivery. Engineers from AREVA organized and followed up these tests. The handling operations were performed by the qualified people on the ANPP site.

On-site support for training purposes was provided during the loading of the first four canisters. After the loading of the last of the first four canisters AREVA established an end-of-commissioning report documenting the qualification and start-up operations. The loading and transfer of the next two canisters were performed by the ANPP with the technical assistance of AREVA engineers. A system operation and maintenance manual, operating procedures and training documents were prepared and issued to Armenian technicians. These documents served as inputs for the preparation of specific documents for the ANPP and to help its teams to perform the loading and transfer of the last five canisters without assistance.

3.4. Involvement of local companies

During the first stage of the construction of the dry storage installation, studies for project civil work were performed by a French company, but the civil work and the construction of the storage modules were carried out by an Armenian company. Some of the components, such as reinforced concrete for the construction of the modules, were also fabricated locally.

4. DRY STORAGE SYSTEM EXTENSION PROJECT

The storage system implemented by AREVA has been in operation since the year 2000. Now the ANPP is in need of an extension of its spent fuel dry storage and the Government of Armenia has decided to construct the second stage of the storage facility.

In 2005, the ANPP signed a new contract with TN International, a subsidiary of AREVA and the new holder of the NUHOMS® licence, to extend its dry storage system by 2 sets of 12 modules. The construction of the new storage will provide the ANPP with the capacity to store only its existing spent fuel.

The new storage is expected to be finished by 2012. The construction of the first set of 12 modules should be carried out in the summer of 2007, with the first canister loading operation planned for the same period.

4.1. Activities included in the extension project

4.1.1. Updating of the safety analysis report

For the dry storage extension project, improvements and amendments will be carried out and integrated in the safety analysis report before its submittal for approval to the ANRA. The safety analysis report will take into account the feedback from the experience of previous operations.

The design of the new storage system will meet the safety requirements imposed by Armenian regulations and follow the general approach of the US 10 CFR 72 rule on Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation.

4.1.2. Engineering studies

The extension of the storage system will include the erection of two sets of 12 modules, while the first stage concerned only one set of 11 modules. Due to this difference, engineering studies for civil work and construction of the modules will be performed to produce detailed technical specifications.

4.1.3. Manufacturing and delivery of equipment

The extension project will also include the manufacturing and delivery of the mechanical parts for NUHOMS® modules, e.g. canisters, canister supports, doors, protection for air inlets and outlet systems, etc. The equipment necessary for each module will be designed on the basis of the already existing 11 spent fuel dry storage systems.

4.1.4. Training programme

A training programme will be organized for the ANPP operators in order to update their competences and qualifications in relation to the process.

4.2. The involvement of the ANPP in the extension project

Erection of modules, assembling of the mechanical internal parts of the modules and canister loading and transfer operations will be performed by the ANPP teams, which acquired the necessary competences and know-how during the first stage of the project. The competences and know-how transferred will allow the ANPP and local companies to get more involved in the second stage of the project. The project will also contribute to the economic and social development of the local communities.

5. CONCLUSIONS

The Armenian spent fuel storage requirements were met by harvesting synergies from the know-how and experience acquired in the same kind of projects carried out in the USA. The ANPP spent fuel storage project demonstrates that NUHOMS® technology can be successfully implemented for fuel systems other than those for which it was originally developed.

REGULATORY ASPECTS OF BUILDING DRY SPENT FUEL STORAGE FACILITIES IN THE RUSSIAN FEDERATION

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Abstract

The experience of applying licensing procedures in relation to the construction of the first dry spent fuel storage facility (XOT-2) in the Russian Federation is described, including the process of review of the licence application, the outcomes of the review process and the implications for the revision of the licence application. Particular attention is paid to issues related to providing assurance of the safety of the planned facility.

1. INTRODUCTION

At the moment, nine nuclear power plants of the WWER-1000 type and RBMK-1000 type are in operation in the Russian Federation (30 units altogether). All spent fuel from RBMK-1000 type reactors is stored in spent fuel pools at the reactors. Spent fuel from the WWER-1000 reactors is kept for 3–5 years in the spent fuel pools of reactors and then transferred to the spent fuel wet storage facility (XOT-1) on the site of the Mining and Chemical Enterprise (MCE) in Zheleznogorsk [1].

The spent fuel pools and the wet fuel storage facility XOT-1 are filled to more than 50% of their capacity. New spent fuel storage facilities are needed or the capacity of the old storage facilities must be increased. Otherwise, it will be necessary to reprocess some part of the stored spent fuel. For the present, priority is being given to constructing dry spent fuel storage facilities.

2. SHORT SPECIFICATION OF THE XOT-2

Construction of the first dry spent fuel storage facility (XOT-2) in the Russian Federation began in 2004 on the site of MCE. The storage facility is intended to be used for the temporary storage of spent fuel from WWER-1000 and RBMK-1000 type reactors.

The site for the construction of the dry storage facility is located on the right bank of the Enisey river at a height of 165–170 m above the water level of the river. The exclusion zone of the XOT-2 will be 1 km and the zone of supervision 30 km. The average population density in the zone of supervision is about 54 persons per square kilometre. The air space above the MCE site is off limits to all types of air transport. There are well developed railway and automobile networks.

The planned storage facility consists of concrete chambers in which there are steel tubes for storing hermetically sealed containers incorporating the spent fuel assemblies. The containers are loaded into the storage tubes in two layers by a reloading machine. As part of the process of long term storage of spent fuel, a procedure for testing the containers on a random basis will be provided. The system of spent fuel storage includes two physical barriers, the hermetic container and the hermetic storage tube. The characteristics of the spent fuel storage are that the duration of the storage is not less than 50 years, the storage medium is nitrogen, the temperature of storage is from 300º to 350º C, the water content is less than 25 g/m^3.

Two stages of XOT-2 commissioning are planned. The first stage will last up to 2012. It is planned to load about 5000 tonnes of spent fuel during this starting phase. The projected capacity of the storage is more then 37 000 t U (including about 24 000 t of RBMK-1000 spent fuel and 9000 t of WWER-1000 spent fuel).

3. LICENSING PROCEDURE

In accordance with Article 26 of the Federal Law on the use of nuclear energy, any activity in the field of nuclear energy use is not allowed without a licence for its execution. The list of types of activities in the field of nuclear energy use that require licensing is established by the Government of the Russian Federation in the Provisions on Licensing of Activity in the Field of Nuclear Energy Use. The Provisions include such activities as siting, construction, operation and decommissioning of a spent fuel storage facility. To obtain a licence an applicant has to submit to the regulatory body, Rostechnadzor, the set of documents defined by the Provisions.

4. EXPERTISE

The most important part of the licensing procedure is the expert examination (review) of the documents proving the safety of the planned facility. The purpose of this review is to obtain an independent assessment of the safety of the activity to be licensed. It is necessary to assess the technical solutions proposed by the applicant on safety assurance and to review them for compliance with the requirements of the rules and regulations in the field of nuclear energy use. It includes compliance with:

(a) Provisions of the Federal Laws on the use of nuclear energy, and other federal laws and regulations;
(b) Special regulatory documents in the field of nuclear energy use;
(c) State of the art technical approaches in the subject area.

The review of the licensing proposal document may be performed by the Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS) or by another expert organization which has a licence issued by Rostechnadzor on its competence to carry out such reviews. The SEC NRS carried out the expert review for the construction of the XOT-2.

The review of the licence application was based on the analysis of more than 70 national documents and some IAEA documents.

An important regulatory document for the safety of a spent fuel storage facility covering the stages of designing, siting, construction, operation and decommissioning is Facility Dry Storage of Spent Fuel: Safety Requirements, which contains specific provisions to provide for the protection of workers, the public and the environment. Another important regulatory document is General Provisions of Safety Assurance Facilities of Nuclear Fuel Cycle. This document sets out the basic principles and criteria for assuring safety and general nuclear and radiation safety requirements for the facilities of the nuclear fuel cycle.

The review took account of the relevant articles of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [2], in particular, articles 4, 6, 7, 8, 23 and 26. These articles contain provisions for the design and construction of spent fuel management facilities. The technical solutions used by the applicant to provide for the safety of radioactive waste management in the context of the XOT-2 were analysed for compliance with the Joint Convention.

The review of the complete set of documents on the safety of the XOT-2 construction phase was carried out in three stages to give the applicant time for the necessary preparatory work. At the first stage of the review (January 2004), the basic problems that should be solved before the start of the XOT-2 construction were examined. Assessments were made of the following:

(1) Conceptual provisions for safety;
(2) Criteria and principles of safety for dry storage;
(3) Influence of external and internal effects of accidents on the reliability of the building structure (of human and natural origin);
(4) Quality assurance programmes.

The main shortcomings and defects revealed by experts at the first stage of the review were remedied by the operating organization by means of measures approved by Rostechnadzor before the construction licence was issued.

The set of main principles for safety assurance provide the basis for the implementation of the dry spent fuel storage project. Ensurance of radiation safety in the project is based on the principles of justification, optimization and dose limitation. The radiation dose limitation criteria for workers and the public are used as a basis for control for the normal operation of the XOT-2 facility and in the case of accidents. The standards for discharge and leakage of radioactive materials into the environment are also used as basic criteria.

One of the most important safety principles used in the basis of the XOT-2 design is the defence in depth concept. The concept is based on using a system of physical barriers to reduce levels of ionizing radiation and to prevent the spread of nuclear materials and radioactive materials in the environment. A system of technical and organizational measures for the protection of the physical barriers and for securing their effectiveness is also used. The main criteria used to judge that this principle has been properly applied are as follows:

(i) Existence of physical barriers;
(ii) Presence of five levels of defence in depth;
(iii) Implementation of the defence in depth concept at all stages of the XOT-2 functioning;
(iv) Obligation of proving the correctness of all technical solutions of facility safety assurance.

The following barriers to prevent the spread of ionizing radiation, nuclear materials and radioactive materials are provided for by the XOT-2 project:

— Fuel matrix;
— Fuel pin casing;

— Hermetic container;
— Hermetic steel socket;
— Reinforced concrete building constructions.

The analysis of the XOT-2 project materials shows that the basic solutions of the project are in compliance with the defence in depth concept. Safety assurance of the XOT-2 is based, first of all, on the principle of not allowing the conditions for starting a spontaneous nuclear chain reaction (nuclear criticality). The criterion for assuring that the systems remain subcritical is that the neutron multiplication factor is less than 0.95. Simulations of the storage conditions for spent fuel from WWER-1000 type reactors and RBMK-1000 type reactors were performed for confirmation of this parameter.

One of the main principles of safety assurance for the safe dry storage of high amounts of spent fuel (140 active zones) for more than 50 years is providing an appropriate temperature regime. A temperature limit of 3000°C was chosen for normal operating conditions and for accident conditions.

Natural convection was chosen as the method for cooling of spent fuel in the XOT-2 storage facility (passive method). The temperature and thermal regimes in the XOT-2 storage cells were calculated with the SINF and CONRAD computer codes. However, the results of the calculations should be regarded as preliminary because the complex thermal non-stationary process of storage needs additional calculations and simulations.

It is necessary to mention that the principles for safety assurance in the XOT-2 project are as follows: prevention of violations of limits and conditions for safe operation, limitation of accident consequences, assignment of the entire responsibility for safety of the facility to the operator.

Minatom of the Russian Federation executes a programme for the scientific support of the XOT-2 project to solve all existing technical problems. The shortcomings identified by experts during the first stage of the review were generally connected with insufficient justification of the technical solutions proposed to guarantee the safety of the XOT-2 operation. The operator eliminated these shortcomings before the licence for the facility construction was issued. The projected decisions that would influence the safety of the XOT-2 operation were examined at the second stage of the review (May 2004). They include assessments of:

(a) Transport technology operations with spent fuel assemblies;
(b) Systems of measurements of spent fuel;
(c) Systems of preparation of spent fuel for its storage;
(d) Systems of long term storage of spent fuel;
(e) Nuclear, radiation, technical and fire safety provision;

(f) Heat elimination system effectiveness;
(g) Radioactive waste management system;
(h) Accounting for and control of nuclear materials and radioactive waste;
(i) Physical protection of the facility;
(j) Plans and procedures for decommissioning of the facility.

All of the expert reviews gave positive conclusions about the possibility of the XOT-2 construction on the site of the Mining and Chemical Enterprise. A positive assessment was given of the concept, principles and criteria for safety assurance for the construction of the XOT-2 dry spent fuel storage facility. Nevertheless, the thorough analysis of the documents presented by the applicant, and especially of the safety analysis report, allowed the experts to find some shortcomings in different aspects of the planned safety measures for the facility.

As a result of the reviews about 140 observations and shortcomings were identified that should be corrected in the project, technical documentation and in the safety analysis report. Among others, it is necessary to mention the following:

(1) Not enough proof was given for the estimates of gas leakage through welded joints of containers and tubes, taking into consideration the fact that with full capacity the XOT-2 will have more then 30 000 welded joints;
(2) The estimate of the probability of guaranteeing the hermetic sealing of containers was performed without taking into account the summed length of welded joints of containers and tubes, and without taking into consideration the processes that will take place inside the structure and construction materials in the course of normal operation or accidents in the period of 50 years;
(3) The absence of a system to provide for periodic operational testing of containers and tubes by air pressure for almost seven thousand sockets;
(4) A possible accident involving transport machinery with destruction of equipment and components inside the storage facility was not considered in the analysis of the initial events;
(5) The method of calculation of the temperature of the elements of tubes and fuel assemblies is inadequate since it is based on a simplified heat transfer mechanism in the air layers between tube walls and the container;
(6) The temperature calculations have not been performed for violations of normal working conditions and accidents;

(7) Some specifications of liquid radioactive waste at different stages of their processing and specifications of the equipment to be used are absent in the project;
(8) Measures for reducing worker exposures during spent fuel transport operations are not considered;
(9) The technology for treating damaged spent fuel assemblies is not described;
(10) The analysis of the possibility of dangerous concentrations of hydrogen being generated is performed with the assumption of a homogeneous distribution in the facility; the consequences of possible local gas accumulation are not assessed.

The regulatory body, Rostechnadzor, issued the licence for construction of the dry storage facility to the applicant in 2004 for a period of 10 years. The licence makes it a condition that the applicant (MCE) must eliminate and take account of all shortcomings and observations that are identified in the reviews made by the SEC NRS, in the resolutions of the Glavgosexpertiza, in the state ecological review, and in the hygienic–epidemiological conclusion provided by 'Medbioextrem' of the Ministry of Health of the Russian Federation.

At the end of 2004, the SEC NRS performed the third stage of the review with the purpose of assessing the actions planned by the applicant and the design organization to eliminate all observations and shortcomings. The results of this review show that the applicant had taken into consideration all of the observations and shortcomings and that they are to be eliminated by 2007.

5. CONCLUSIONS

At present, according to the licence terms, the applicant (MCE) is working on clarifications and elaborations of some of the project decisions, such as:

(a) Influence of an earthquake;
(b) Control of hermetically sealed physical barriers;
(c) Justification of possible gas escape from a storage tube;
(d) Improvement of radioactive waste management;
(e) Management of leakage from fuel and other sources.

Inspectors of the regional department of Rostechnadzor in Zheleznogorsk supervise compliance with the licence terms, the construction works and the elimination of shortcomings.

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REGULATORY BODY EXPERIENCES IN LICENSING AND INSPECTION OF DRY CASK STORAGE FACILITIES

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Abstract

The Nuclear Regulatory Commission (NRC), through a rigorous licensing and inspection programme, ensures the safety and security of dry cask storage. The NRC authorizes the storage of spent fuel at an independent spent fuel storage installation (ISFSI) under two licensing options: site specific licensing and general licensing. In July 1986, the NRC issued the first site specific licence to the Surry Nuclear Power Plant in Virginia, authorizing the interim storage of spent fuel in a dry storage cask configuration. Presently, there are over 40 ISFSIs licensed by the NRC, with over 800 loaded dry casks. Current projections indicate that there will be over 50 ISFSIs by the year 2010. No releases of spent fuel dry storage cask contents or other significant safety problems from the storage systems in use today have been reported. The paper discusses the NRC's licensing and inspection experiences.

1. INTRODUCTION

1.1. Background

All operating nuclear power reactors in the USA are storing spent fuel in Nuclear Regulatory Commission (NRC) licensed on-site spent fuel pools (SFPs). Most reactor pools were not designed to accommodate the full amount of spent fuel generated during the lifetime of the reactor. Utilities originally planned for spent fuel to remain in the SFPs for a few years after discharge from the reactor core and then to be sent to a reprocessing facility. However, the US Government declared a moratorium on reprocessing in 1977. Although the ban was later lifted, reprocessing has not been pursued since then. Consequently, utilities expanded the storage capacity of SFPs by the use of high density storage racks. Eventually, utilities needed additional storage capacity. In response to these needs, the NRC provided a regulatory alternative: interim spent fuel storage in dry cask storage systems. For spent fuel management both pool storage and dry storage are safe methods, but there are significant differences between them. Pool storage requires a greater degree of operational control on the part of the nuclear power plant operators — to maintain the performance of the electrical and mechanical systems supporting pumps, piping and instrumentation. Dry storage technology uses passive cooling systems with robust cask designs requiring minimal operational controls.

1.2. Site specific licence

Initially, the NRC only authorized the storage of spent fuel at an independent spent fuel storage installation (ISFSI) by means of a site specific licence under the authority of the Atomic Energy Act of 1954, as amended. Under a site specific licence, an applicant submits a licence application to the NRC. The NRC performs a technical review of all the safety, security, and environmental aspects of the proposed storage facility. An opportunity for public involvement is provided through a formal hearing process administered by the independent NRC Atomic Safety Licensing Board. If the application is approved the NRC issues an ISFSI licence valid for 20 years. The NRC regulations also include provisions for the renewal of an ISFSI licence. A spent fuel storage licence contains technical requirements and operating conditions for the ISFSI, and very specific conditions on the spent fuel that the licensee is authorized to store at the site.

The Nuclear Waste Policy Act of 1982, as amended (NWPA) acknowledged the need for new storage capacity at the site of each civilian nuclear power reactor [1]. The NWPA directed the NRC to take such actions as it considered necessary to encourage and expedite the effective use of available storage and necessary additional interim storage. In response, the NRC amended its regulations to provide a second licensing option (referred to as the general licence) for the storage of spent fuel.

1.3. General licence

The NRC regulations provide a general licence which allows nuclear power reactor licensees to store spent fuel in dry storage systems approved by the NRC at a site that is already licensed to operate a nuclear power reactor under 10 CFR Part 50. A wide variety of dry storage systems have already been approved by the NRC for general licensees to consider. Fifteen dry storage designs have received certificates of compliance (CoC) and are listed in NRC regulations [4] (Part 72.214). General licensees are required to perform evaluations of their sites to demonstrate that the sites are adequate for storing spent fuel in dry casks. These evaluations must show that the CoC conditions and technical specifications can be met prior to the use of the selected dry storage system at the general licensee's site.

The NRC approves spent fuel dry cask storage systems by evaluating each design for a wide range of normal conditions of use and accident conditions such as floods, earthquakes, tornados, missiles, and temperature extremes. The NRC issues a CoC for a cask design if the review of the design finds it technically adequate. The cask certificate expires 20 years after the date of issue with an option for subsequent re-approval. The NRC has prepared standard review plans to guide the NRC staff review of both site specific ISFSI license applications and dry cask storage system applications (NUREG-1567 and NUREG-1536, respectively) [2, 3], which are available on the NRC Web page, http://www.nrc.gov/waste/spent-fuel-storage.html [6].

1.4. Inspection

The NRC conducts periodic inspections of both site specific licensees and general licensees. Inspections occur during cask design, cask construction, site preparation and ISFSI pad construction, pre-operational or trial run, cask loading, and ISFSI operations. The inspections are conducted at cask design facilities, cask manufacturing plants, and the ISFSI site. The inspection programme has the same underlying safety and security focus for facilities licensed under either a site specific or general licensing option. The NRC has prepared procedures to guide NRC staff on the inspection of both site specific ISFSI licence applications and dry cask storage system applications (MC 2690, MC 2690A and MC 2690B), which are available on the NRC web page [5, 6].

2. DISCUSSION

The NRC's strategic plan presents its mission, vision, goals and outcomes that will guide its strategic direction for the next 5 years. The strategic plan is centred around five goals; safety, security, openness, effectiveness and management. The first and most important goal is safety. The strategic plan goes into detail on how NRC's storage activities feature prominently in strategies for accomplishing these goals. For example, the completion of technical reviews of spent fuel dry storage systems to ensure that they will be safe and secure for use at any licensed spent fuel storage facility is specifically tracked. Achieving these strategic goals requires the collective effort of the NRC and its licensees and certificate holders.

Currently, there are over 40 ISFSIs licensed by the NRC with over 800 loaded dry casks in 26 states. There has been a marked increase in the rate of new licensees over the past few years. For example, the number of ISFSIs has more than tripled from 12 in 1999 to the present. Additional site specific ISFSIs and generally licensed ISFSIs are expected to be operating within the next few years with current projections of over 50 ISFSIs by the year 2010. The workload for cask certifications will continue to grow as nuclear power plants require increased spent fuel storage capacity and as designs are proposed that accommodate higher burnup fuel and higher heat loads for recently discharged spent fuel.

Openness, involving communication and involvement between the NRC and its stakeholders, is the cornerstone of strong, fair regulations and is key to maintaining and improving safety performance. The NRC held a ‘first of a kind’ engagement with industry stakeholders at a licensing process review conference in February 2005 to solicit feedback, including suggestions for improvements, on its performance and recommendations. Over 140 representatives of the regulated industry, other Federal agencies, cask designers, media and press, and members of the public participated in a full day meeting at which experiences were discussed and suggestions made for process improvements. The NRC has adopted many of the suggestions identified during the conference, including the establishment of an industry–NRC task force to focus on process improvement. Two examples of initiatives being addressed are: improving the process for requests for additional information, and the issue of interim staff guidance documents (ISGs). Standard review plans have been augmented with ISGs to support timely decisions about technical and regulatory issues. Currently, 22 ISGs have been issued and the process has changed to include stakeholder input and review of draft ISGs to ensure that the final ISG is reflective of stakeholder experience. To address emerging technical needs, the NRC is considering future ISGs for high burnup fuel, defining thermal analysis parameters, clarifying the definition of damaged fuel and fuel retrievability, and allowance for additional burnup credit. Resolution of these issues is needed to ensure that future designs have an adequate technical basis to comply with regulatory decisions and to meet industry spent fuel storage needs.

An example of how the NRC is increasing the efficiency of its regulations is its initiative to provide increased licence renewal terms for ISFSI licensees. The NRC recently issued a licence renewal for the Surry Nuclear Power Plant ISFSI for a 40 year ISFSI licence renewal term. The regulations permit a 20 year licence renewal term, but the licensee requested an exemption to allow a 40 year term and provided a technical basis that NRC staff, in their review, found acceptable. The Surry ISFSI was the first dry cask storage site licensed in the USA, in 1986, and is the holder of the first licence renewal granted for an ISFSI. Following the decision, the Commission directed the NRC staff to explore a potential rule to permanently change the licence duration set in 10 CFR Part 72 [4]. Since then, the H.B. Robinson nuclear power plant ISFSI has also been granted an exemption, and a 40 year renewal period has been approved for its ISFSI licence.

The NRC has many security challenges to face as a result of a changing terrorist environment that poses potential threats to the security of nuclear installations. The NRC is actively evaluating potential security challenges to spent fuel storage facilities and operations, as well as to the transport of spent fuel, to ensure that the enhanced security programmes address these new challenges. Additionally, the NRC has issued several licence orders to supplement the security requirements in 10 CFR Part 73 for spent fuel storage and transport. This is a dynamic area and the NRC continues to evaluate the threat environment and licensee performance to determine if additional security measures are needed.

Regarding security assessments, the NRC has evaluated the impact on spent fuel storage casks and transport packages of various hypothetical terrorist events, including large plane crashes into interim storage facilities, an attack on a spent fuel transport package, and other acts of terrorism. These studies evaluated both un-reinforced and reinforced concrete steel spent fuel storage casks with bolted and welded lid closures and horizontal storage in reinforced concrete. The studies also evaluated impacts on rail and legal weight truck spent fuel transport packages and various consignment packages of radioactive material. These studies were completed in 2005 and the NRC is considering the staff recommendations based on the outcome of these evaluations.

The NRC conducts approximately 20 inspections of licensees/vendors/fabricators each year. The NRC uses a systematic approach to identify facilities and sites for inspection considering, for example, the organization's inspection history, reported events, and level of design and fabrication activity. The NRC inspects both domestic and foreign vendors and fabricators. The NRC inspection activities have increased to reflect the growth in ISFSIs and the increased fabrication demands to supply casks for ISFSIs. The NRC inspections have resulted in valuable experience that has been, and continues to be, incorporated into the inspection programme, as well as lessons learned that the NRC has passed on to the industry. Pre-operational trial run inspections are very useful to both the licensee and the NRC to demonstrate that the licensee is ready to load fuel into the cask and has done adequate site evaluation and preparation. The NRC inspections have also focused licensee attention on the need to maintain a questioning attitude, examining changes in procedure for new potential problems, and the importance of an active quality assurance oversight programme for all processes of the cask vendor and cask fabricator operations to ensure the quality of the final products. Another valuable lesson learned from the NRC inspections is the need for licensees to perform early characterization of the fuel to be moved to the cask to ensure that it meets the conditions of the dry cask storage system CoC.

Recently, integrating technical review staff with inspection staff as an element of NRC's knowledge management programme has resulted in an improvement in inspections, technical evaluations and licensing documents which are more usable by inspectors.

As the storage technology developed, both the NRC and industry learned how to apply NRC regulations and how to conduct inspections. For example, lessons learned included the need to properly test multiple pass root welds, the need to consider heavy loads in relation to floor loading, that some coatings generate hydrogen gas faster than others, and that quality assurance during the fabrication of cask storage facilities and cask structures must be augmented by close oversight by the utility and cask vendor. The NRC continues to conduct its licensing and inspection programme with the flexibility to adjust based on operational experience.

Public interest in the safety and security of nuclear facilities and the storage of nuclear waste has grown, and is expected to continue to increase, as a result of the increasing demand for nuclear power plant licence renewals and the prospect of new plant construction. These activities have led to an increase in the public scrutiny of the national nuclear waste policy, namely, of how to safely store the growing inventory of spent nuclear fuel and ultimately, how to dispose of it.

3. CONCLUSIONS

The NRC is committed to furthering its strategic goals by ensuring effectiveness and efficiencies through continuous improvement of its processes and management actions, and to actively engage and inform its stakeholders of its activities. The NRC regulates the civilian use of byproduct, source and special nuclear materials to ensure adequate protection of public health and safety, promote the common defence and security, and protect the environment. To support this mission, the Spent Fuel Project Office of the NRC enables the safe interim storage of spent fuel. This is accomplished through its regulatory oversight, licensing, inspection and technical review of ISFSIs, as well as the issuance of CoCs.

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STORAGE CONTAINERS

(Session 5)

Chairperson

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LICENSING OF THE CASTOR© 440/84M CASK FOR TRANSPORT AND STORAGE OF SPENT FUEL

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Abstract

Dry cask storage technology has been used in the Czech Republic since December 1995, when trial operation of the Interim Spent Fuel Storage Facility (ISFSF) at the Dukovany NPP site started. Storage is based on the use of the CASTOR© 440/84 cask. As the storage capacity of the Dukovany ISFSF is limited to 600 t HM (in 60 CASTOR© 440/84 casks), in the late 1990s it was decided to launch activities related to the design, construction and operation of a new spent fuel storage facility at the same site (SFSF Dukovany). The new facility will use a modified CASTOR© 440/84M cask. The paper describes the experience gained in the licensing of the new cask for the transport and storage of spent fuel.

1. INTRODUCTION

Dry cask storage technology for the temporary storage of spent fuel from power reactors has been used in the Czech Republic since December 1995, when the trial operation of the Interim Spent Fuel Storage Facility at the Dukovany NPP site (the Dukovany ISFSF) was started. As the storage capacity of the Dukovany ISFSF is limited (fully used up by early 2006) and the preferred national policy on radioactive waste and spent fuel management involves dry cask spent fuel storage at the sites of nuclear power plants (NPPs), it became obvious that the construction of a new spent fuel storage facility at the Dukovany site (Dukovany SFSF) was needed. The storage capacity of the Dukovany SFSF (1340 t HM) will be sufficient for the storage of all spent fuel from the Dukovany nuclear power plant once the Dukovany SFSF is full and until the decommissioning of all four units of the Dukovany nuclear power plant.

The siting of the Dukovany SFSF was started almost in parallel with the assessment of its environmental impact in the summer of 1998. In parallel to the siting procedure, the future Dukovany SFSF operator started a process for the selection of the cask supplier for the initial operation of the Dukovany SFSF. Based on previous experience with the CASTOR© 440/84 cask, a

German company, GNB mbH (now GNS mbH), was selected as the cask supplier for the initial period of the Dukovany SFSF's operation. The cask, which will be used in the Dukovany SFSF, is the CASTOR© 440/84M (M for modified) cask.

In summary, the following key steps have been taken in relation to the construction of the Dukovany SFSF:

(a) Nuclear safety is a global issue rather than only a national issue;
(b) Development of environmental impact documents for the Dukovany SFSF;
(c) Receipt of an expert opinion on the environmental impact documents for the Dukovany SFSF;
(d) Holding of a public discussion on the environmental impacts of the Dukovany SFSF;
(e) Issue of a favourable position by the Ministry of Environment;
(f) Development of the initial safety report in connection with the application for a siting licence;
(g) Issuing of a siting licence by the State Office for Nuclear Safety (SÚJB) for the Dukovany SFSF at the NPP Dukovany site (December 1999);
(h) Issuing of a planning permit;
(i) Selection of a designer for the Dukovany SFSF;
(j) Selection of a cask supplier for initial operation of the Dukovany SFSF;
(k) Development of a preliminary safety report in connection with the application for a license for construction of the Dukovany SFSF;
(l) Issue of a licence by SÚJB to construct the Dukovany SFSF at the Dukovany site (October 2002);
(m) Start of construction of the Dukovany SFSF (April 2004);
(n) Licensing of the CASTOR© 440/84M cask (July 2005);
(o) End of construction and facility inspection by the responsible regional construction office (February 2005);
(p) Beginning of the commissioning process (March 2006).

2. CASK LICENSING PROCEDURE

The CASTOR© 440/84M cask type approval (licensing) procedure was launched on 23 June 2003 when the cask designer, the German based GNB mbH (now GNS mbH) company, submitted an official application for cask licensing to the national regulatory body, the State Office for Nuclear Safety (SÚJB). The cask licensing procedure followed the requirements of:

FIG. 1. The Dukovany ISFSF and SFSF.

(a) The national regulations as defined especially in Act No. 18/1997 Coll. on peaceful utilization of nuclear energy and ionizing radiation and on amendments to and alterations of some acts (Atomic Act) and Decree No. 317/2002 Coll., on type approval of packaging for transport, storage and disposal of nuclear materials and radioactive substances, on type approval of ionizing radiation sources and transport of nuclear materials and specified radioactive substances (on type approval and transport);
(b) The IAEA requirements published in Safety Standards Series No. TS-R-1, Regulations for the Safe Transport of Radioactive Material, 1996 Edition (As Amended 2003), IAEA, Vienna, 2004.

The submitted request contained the cask safety case documentation [1] and other documents required by the national regulations. Within the framework of the cask licensing procedures, several independent review reports were also submitted to the SÚJB. They were related to the calculation of rod cladding temperatures during the cask drying [2], cask inventory, criticality calculation, confinement system and thermal calculation assessment [3], assessment of cask shielding properties [4] and cask mechanical strength evaluation [5].

The licensing procedure was finished after about 2 years when SÚJB issued the licence for the CASTOR® 440/84M cask for railway transport and storage of spent fuel. The licence contains the conditions under which the cask can be used. They cover items such as:

(1) Cask inventory (number and type of loaded fuel assemblies, total cask inventory and thermal output);

(2) Cask loading pattern (homogeneous and heterogeneous loading according to the burnup and cooling time and according to the inventory of the fuel assemblies and γ/n radiation intensity);
(3) Cask cooling (by helium);
(4) Drying criteria for the loaded cask;
(5) Steady state temperature;
(6) Leaktightness tests;
(7) Decontamination and dosimetric control;
(8) Maximum cask surface temperature;
(9) Maintenance and handling;
(10) Operational controls;
(11) Quality assurance;
(12) Cask markings;
(13) General requirements (licensees and manufacturers responsibilities, additional conditions on the issue of new cask licence, cask conformity reporting, etc.);
(14) Emergency notification.

The license for the CASTOR® 440/84M cask was issued for a limited time period — until mid-2010. This corresponds to the general practice in the Czech Republic. Before the licence's expiration the licensee has to submit to the SÚJB a new application for a cask licence, taking account the operational experiences obtained, with emphasis on the ageing of cask material and subsystems.

3. CASK SAFETY CASE

The CASTOR© 440/84M cask design is based on the design of the CASTOR© 440/84 cask. However, several important modifications can be identified in the design of the new cask related mainly to the:

(a) Improved neutron shielding properties of the cask;
(b) Modified trunnion construction;
(c) Optional use of a third welded lid;
(d) New design of the fuel basket.

The cask can be operated in three basic configurations, transport (with two lids and shock absorbers), storage (with two lids and protective plate only; see Fig. 2) and storage with a third lid.

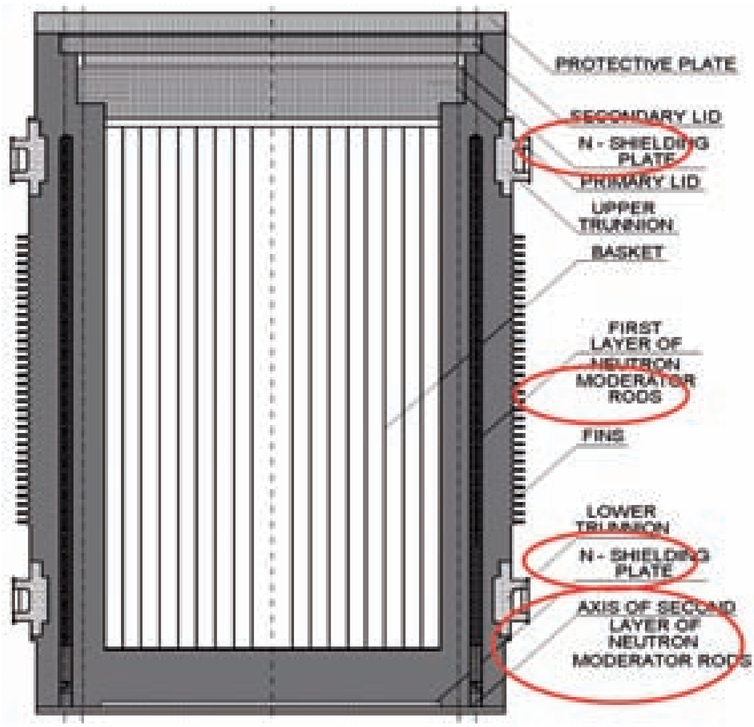

FIG. 2. Storage configuration of the CASTOR© 440/84 M cask.

In addition, the operational procedures for the modified cask differ from those for the older CASTOR© 440/84 cask in several aspects. The changes are related, for example, to the increased requirements on primary and secondary lid leaktightness (from 10^{-7} to 10^{-8} hPa·L/s) and to the modified cask drying procedures, with a more restrictive gas pressure change criterion defining the level of cask drying (Δp from 0.3 to 0.2 hPa/15 min). New procedures were developed for cask drying for higher spent fuel thermal loads (>16 kW), for fuel integrity measurement with the help of AAM/AS monitoring equipment, for the installation of a third welded lid, etc. All operational procedures will be subject to operational verification tests during commissioning of the Dukovany SFSF.

The cask safety case contains three main parts: project (mainly a description of the cask components, safety assessment and evaluation of results), construction (scale drawings and list of cask components) and operation/maintenance (operation and maintenance procedures). In the following part of this section the main conclusions of the cask safety assessment are described.

3.1. Cask criticality calculation

The cask criticality calculation is based on the conservative initial assumption that loaded fuel has the physical properties of fresh fuel, i.e. the burnup is not used in the criticality calculation. The average initial enrichment of the profiled fuel assembly is 3.82% wt of ^{235}U. The computer codes MCNP4B and WIMS8 were used for 3-D calculations of k_{eff} and (2-D) for sensitivity analysis, respectively.

The reference criticality calculation was performed for optimal moderation, i.e. for a water environment without any additives, for profiled fuel and using the detailed properties of the cask body and basket. Additional calculations for emergency situations (a fall from 9 m with consecutive cask flooding) involving basket deformation and fuel assembly squeezing of 3 mm, showed that the internal deformations of the basket have the predominant effect on the calculated value of k_{eff} and that the fuel assembly squeezing effect reduces the value of k_{eff}.

The calculated k_{eff} for normal operation, including a systematic calculation deviation (0.0085), material property deviations (0.0052), a temperature correction (0.003), a standard deviation (0.0006) and a methodology bias (0.0048), is equal to 0.93022, which is 1.9% less then regulatory limit of 0.95. For the emergency situation, taking into account the mechanical deformation of the basket (0.001) and gradual cask flooding (0.001), the value of k_{eff} equals 0.93222.

3.2. Cask inventory and shielding calculation

The CASTOR© 440/84 M cask is designed for the transport and storage of 84 WWER-440 type fuel assemblies. The basic physical properties of the loaded spent fuel are:

(a)	Thermal output of all loaded assemblies	max. 24 66 kW (290–340 w per assembly)
(b)	Radioactivity of all loaded assemblies	max. 2.6 × 017 Bq
(c)	Burnup of one fuel assembly	max. 50 GW·d/t
(d)	Initial fresh fuel enrichment	max. 3.87 wt% of ^{235}U

The radioactive inventory and γ/n radiation dose calculations were performed using the ORIGEN-2.1 computer code using the PWRUS and PWRUE libraries of reaction effective cross-sections. Six initial conditions of spent fuel were considered. For these six combinations of spent fuel properties the activities and radiation doses were determined.

TABLE 1. BASIC PHYSICAL PROPERTIES OF SPENT FUEL ASSEMBLIES

Parameter	Unit	Fuel assembly type					
		A-1	A-2	A-3	A-4	A-5	A-6
A.1. Max. initial enrichment	wt% of ^{235}U	1.60	2.35	3.55	3.77	3.55	3.77
Max. burnup	GW·d/tU	25	35	42	45	48	50
Min. cooling time	month	72	72	72	72	90	90
Max. thermal output	W/assembly	290	290	290	290	290	290

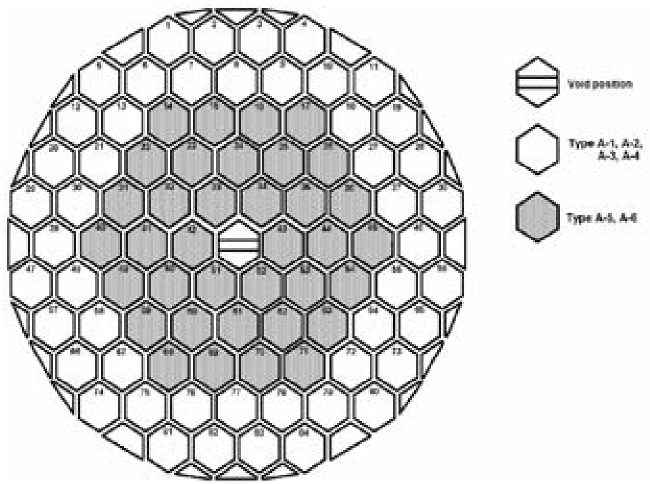

FIG. 3. Heterogeneous cask loading pattern.

The results of inventory calculations were used for the determination of the cask loading patterns and for the assessment of the cask shielding properties. The cask can be loaded according to the basic physical properties of the spent fuel (see Table 1) in two ways — homogeneously (an arbitrary combination of spent fuel types A-1 to A-4), or heterogeneously. In the latter case, based on a zoning approach, there are fuel assemblies of type A-5 and A-6 in the central positions, and in the remaining void positions, assemblies of type A-1 to A-4 (see Fig. 3).

Additionally, the cask can be heterogeneously loaded according to the level of the γ/n radiation and inventory of radionuclides. In this case the original six type categorization is reduced to three types (B-1 to B-3). In addition, six fuel assemblies with a thermal output of up to 340 W (B-3 type) can also be loaded into the cask.

The radiation dose rate simulation on the cask surface and at a defined distance from the cask surface is achieved with the help of the MCNP-4C computer code using a Monte Carlo simulation method. From the initial γ/n radiation intensity for six types of fuel assembly, reference fuel assemblies with the highest γ and n radiation levels were selected (A-4 and A-6). By calculation, the activity of the upper and lower parts of the fuel assemblies containing only activated construction materials and the axial distribution of radiation intensity in the fuel zone of the assemblies (peaking) were considered. The calculation models correspond to both transport and storage configurations. For each configuration, two calculational models were developed — one set for normal transport and storage conditions and the second set for accident conditions when all neutron shielding materials and the secondary lid were assumed to be absent. For the transport model, under normal and accident conditions, the presence of shock absorbers was also omitted.

From Fig. 4 it is obvious that the main cask surface dose rate component (72%) is related to the neutron radiation from the reference spent fuel assemblies. For accident conditions by transport, when neutron moderation is not available, the neutron radiation component exceeds 90%. Similar results were obtained for the storage configuration.

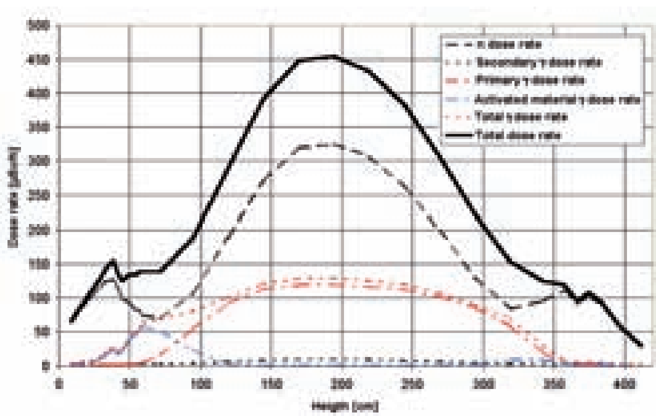

FIG. 4. Cask surface dose rates (transport configuration, normal operating conditions)

3.3. Thermal load assessment

Several sets of thermal load calculations were performed within the framework of the cask safety case. The goal was to determine the fuel cladding and cask surface (and other construction components) temperature for transport and storage configurations under normal and accident conditions and to evaluate the fuel cladding temperatures during loaded cask drying procedures.

All calculations were performed with the COSMOS/M computer code. The initial conditions for transport under normal and accident conditions (fire, 800°C for 30 minutes) were taken from Decree No. 317/2002 Coll., on type approval and shipment, which adopts IAEA Safety Standards Series No. TS-R-1, Regulations for the Safe Transport of Radioactive Material, into the Czech national regulatory system. Both homogeneous and heterogeneous cask loadings were considered in the thermal field simulations, together with spent fuel peaking (peaking factor up to 1.156). The simulation for normal operation was performed under stationary conditions and for accidents under transient conditions. The fuel cladding temperature did not exceed the temperature limit of 350°C and the cask surface temperature did not exceed 85°C under normal conditions.

The simulation of the storage configuration followed the methodology used for the transport configuration. However, the scope of the accident condition simulation contained not only fire, but also the collapse of the storage facility, leading to the restriction of heat removal by convection. Neither under normal conditions nor for fire conditions did the cladding temperature reach the 350°C limit. On the other hand, for the collapse of the storage facility the cladding temperature reached 350°C after about 85 hours. This result will be used for the definition of emergency procedures for the Dukovany SFSF.

The third set of thermal calculations simulated the change of fuel and cask component temperatures during the cask drying at the service place in the reactor hall. For different thermal loads, reaching the maximum value of 24.66 kW and for different contents of helium in the helium vapour mixture, transient simulations of the fuel cladding temperature were performed. At the maximum residual fuel power in the cask, the maximum fuel cladding temperature of 330(±20)°C during the cask drying was reached in about 27.5 hours. If the thermal load of the cask does not exceed 17.1 kW, the fuel cladding at the stationary state ($t \to \infty$) is below 330(±20)°C (see Fig. 5). The limiting volume share of helium in the mixture of helium vapour in the free cask space during the cask drying is 41.4% of helium. At this volume share of helium, the fuel cladding temperature limit of 330(±20)°C is just reached in the stationary state.

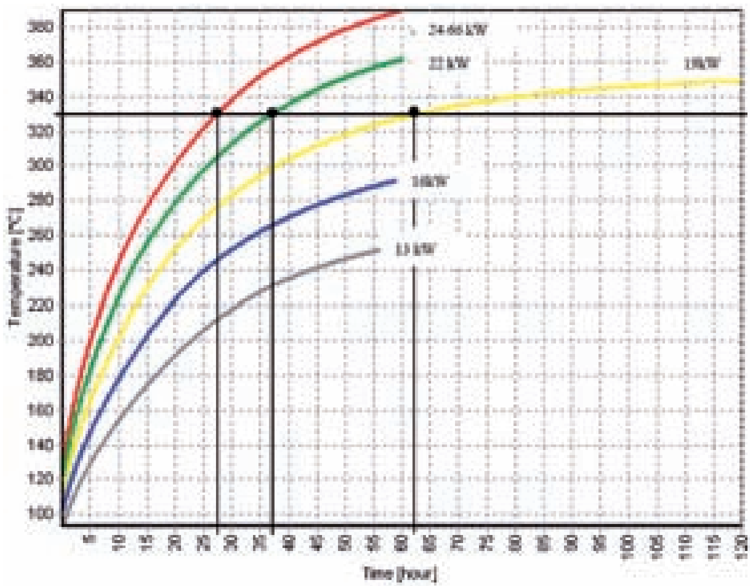

FIG. 5. Temperature of maximum fuel cladding during the cask drying (no helium in the mixture).

3.4. Confinement system

The confinement system of the CASTOR© 440/84 cask isolates the radioactive content of the cask with the help of metallic O rings with stainless steel and aluminium shells and a spring core (on lids) and the cask body itself. Under normal transport conditions the loss of radioactive content cannot be more than 10^{-6} A2 per hour and under accident conditions the accumulated loss of radioactive content in a period of one week cannot exceed 10 A2 for krypton-85 and A2 for all other radionuclides (as defined in Decree No. 317/2002 Coll.). For all calculations of the release of radioactive content, the following radionuclides in the gaseous and volatile phase were considered: ^{3}H, ^{85}Kr, ^{129}I, ^{134}Cs and ^{137}Cs.

Under normal transport conditions, the He leak rate from each of the lids is less than 10^{-7} hPa·L/s and under accident conditions based on experimental data for secondary lid, less than 5×10^{-5} hPa·L/s. It is assumed that 1% of the loaded fuel assemblies are not hermetically tight and that this ratio increases to 100% under accident conditions. About 10% of the total inventory of radioactive gases is not bound to the fuel assemblies and can escape into the cask. The temperatures of sealing and other components of the confinement system were calculated in a previous step of the safety assessment, in the thermal load analysis. Three mathematical models were used for the calculation of release, molecular flow, laminar flow and combined molecular and laminar flow described by the Knudsen equation. For 3-H, the maximum release rate is achieved by the application of the molecular flow model (34.8 Bq/h) and for ^{85}Kr by the Knudsen model (240 Bq/h). These values are several orders of magnitude below the regulatory limits defined in Ref. [6] (4 × 107 and 108 Bq/h). For the accident transport conditions, conservative values are derived from the Knudsen model and are 1.07 × 109 and 4.58 × 1010 Bq/week (the regulatory limit values are 4 × 1013 and 1014 Bq/week, respectively). Due to the high reactivity of radiocaesium in the cask and the unlimited value of A2 for ^{129}I, the calculations for the remaining radionuclides were not performed.

For long term storage conditions it is assumed that due to, for example, the mechanical loads caused by accidents, up to 10% of fuel assemblies are not hermetically tight. Experimental values of the relative release of radioactive gases from stored fuel assemblies used in German power plants [7–10] showed that they do not exceed 6–7% for ^{85}Kr, 0.006% for ^{129}I and 0.9% for ^{3}H. Therefore, conservative values of 10%, 0.01% and 1% for these nuclides were used. Release of ^{137}Cs was not observed in these experiments. However, Cs was included in the calculation in the chemical form CsI. For a two barrier system a molecular flow model was used. Table 2 shows calculated maximum release rates and total release rates from all 133 casks stored at the Dukovany SFSF. Conservatively, it is considered that maximum release occurs at the same time for all casks. The results are compared with regulatory limits for transport as defined in Ref. [6]. The calculated values are several orders of magnitude lower than the transport regulatory release limits which are, according to Ref. [6], also applicable to storage conditions.

TABLE 2. RESULTS OF CONFINEMENT SYSTEM ASSESSMENT FOR STORAGE CONDITIONS

Nuclide	Maximum release rate at time		Total release rate from all 133 casks		Regulatory limit
	(Bq/a)	(a)	(Bq/a)	(Bq/h)	(Bq/h)
^{3}H	2.37×0^{3}	16	3.15×0^{5}	36	4×0^{7}
^{85}Kr	943	14	1.25×0^{5}	14,3	1×0^{7}
^{129}I	9.77×0^{-4}	60	0,13	45×0^{-5}	unlimited
^{134}Cs	82.5	3	1.10×0^{4}	1.25	7×0^{5}
^{137}Cs	145	40	1.93×0^{4}	2.2	6×0^{5}

3.5. Mechanical load assessment

The assessments for cask transport configuration were performed with the ANSYS/LS-DYNA computer code and, in the case of the 9 m and 1 m drop tests, they were recalculated from the results of scale tests under the conditions described in Ref. [6]. The projected accidents for the storage configuration cover:

(a) Gas explosion;
(b) Earthquake;
(c) Impact of displaced cask on other casks stored in the storage facility (domino effect);
(d) Drop from 0.3 m as a result of accident involving the manipulation of the cask;
(e) Impact of a small aircraft engine on the cask body and lids as a result of an aircraft crash.

In the case of a gas explosion (45 kPa), the simulation showed that the cask will not be turned over and its maximum side displacement will not exceed 23 mm. Similar results were obtained for the evaluation of the earthquake impact on cask stability. For an earthquake producing acceleration values of ±0.2 g in the horizontal and ±0.1 g in the vertical direction the cask remains stable. When the cask in the storage facility is displaced as a result of the action of the crane with a horizontal speed of less than 0.33 m/s, the kinetic energy of the moving cask is not sufficient to cause a domino effect on the other stored casks. The mechanical integrity of a cask exposed to a mechanical load by a drop from 0.3 m was simulated with the help of a finite element MKP-LS-DYNA computer code and the results were compared with the results of a similar simulation of a drop test from a height of 9 m. The cask configuration was without the shock absorbers. The calculated value of deceleration, 60g, is significantly smaller than the value of 95 g obtained by the simulation of the 9 m drop test. The last set of computer simulations evaluated the impact of an aircraft crash on the storage facility, focusing on the mechanical load on cask components due to a direct hit by the engine with a weight of 300 kg and a speed of 80 m/s. The maximum displacement between the primary lid and the cask body is less than 0.13 mm and, after vibration damping, less than 25 μm (the elasticity of the sealing can handle a displacement of up to 230 μm).

4. FUTURE ACTIVITIES

The Dukovany SFSF is currently in the commissioning phase. As a part of the commissioning programme, all operational and maintenance procedures for the CASTOR© 440/84M cask will be tested and, if necessary, modified with the direct support of the cask supplier. Operational procedures for AAM/AS monitoring equipment will be developed and tested. This equipment will be used for monitoring the integrity of fuel loaded into casks. It allows not only the detection of any release of radioactive material from fuel cladding to the cask shaft, but also the identification of a leaky assembly or assemblies. The AAM system monitors the vapours pumped out of the cask shaft during the vacuum drying procedure. With the help of a HPGe semiconductor detector and plastic scintillator detector it allows the measurement of the gamma activity of aerosols (trapped in filters) and the beta and gamma activity of noble gases (^{85}Kr, ^{133}Xe). After a leaky fuel assembly or assemblies are detected, the AS hydraulic unit, together with the AAM system, are used for their precise identification. The cask is submerged in a manipulation shaft and the AS unit fixed to the loading machine. The AS unit pumps water from a single loaded fuel assembly into a separator, from which released gases (if any) are transported into the noble gas plastic scintillator of the AAM system.

As a part of the commissioning licence, new operational limits and conditions for the Dukovany SFSF have to be defined and approved by the regulatory body. Generally, they consist of limiting parameters, limiting values, system limiting conditions and administrative conditions. The limiting parameters are related to, in particular, and as appropriate:

(a) Environmental conditions within the store (e.g. temperature, humidity…);
(b) The effects of heat generation from the spent fuel, both for each individual cask (based on cask safety case) and the whole store (e.g. maximum thermal output of the spent fuel assembly/whole cask);
(c) Criticality prevention, covering both individual packages (based on the cask safety case) and the whole store (including operational and accident conditions);
(d) Radiation protection of operating staff and members of a critical group of the public.

Limiting values are values of limiting parameters which must be respected for adequate control of the operation. They are values that must not be exceeded in order to protect the integrity of the physical system designed to prevent the uncontrolled release of radionuclides. The system limiting conditions define the conditions of safety relevant structures and components of the storage facility so that the facility is operated in accordance with the design assumptions and intent, e.g. use of handling equipment for the manipulation of heavy loads, use of licensed casks only, etc. These conditions cover the ventilation system, the fire protection system and the surveillance system (monitoring, inspection and calibration). Administrative conditions are required to ensure that operating, emergency and management procedures, record keeping, review and audit and the transfer and storage of spent fuel in storage are performed in a safe manner.

It is expected that the Dukovany SFSF commissioning phase will last until about the end of 2007. At that time, about 4–6 CASTOR© 440/84M casks will be placed into storage. After this, the regulatory process for issuance of the operating licence will begin.

The experience gained during the CASTOR© 440/84M cask and the Dukovany SFSF licensing will be used for the licensing of a cask and storage facility at the site of the second NPP in the Czech Republic, Temelín. In view of the capacity of the spent fuel pools at the Temelín NPP it will be necessary to commission the storage facility at the Temelín NPP by 2014 at the latest.

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ANTICIPATING FUTURE NEEDS FOR THE TRANSPORT AND STORAGE OF SPENT FUEL FROM EVOLUTIONARY NUCLEAR POWER REACTORS

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Abstract

New materials for use in spent fuel packaging are needed to cope with the technical challenges posed by the highly enriched and high burnup spent fuels expected from evolutionary nuclear power reactors, and also from existing reactors. Research and development undertaken on materials for this purpose is described in the paper and examples are given of the successful application of the new materials.

1. INTRODUCTION

A continuous improvement in fuel design is associated with the increase of nuclear power reactor performance; this is notably the case for evolutionary reactors such as the European pressurised water reactor (EPR), but also for existing reactors. Competitiveness in the nuclear industry relies on, amongst other things, optimal fuel utilization, including the ability for quick removal of fuel assemblies from the cores. At the same time, these new fuel designs show enhanced safety features.

These evolutions lead to new challenges and, in particular, the management of spent fuel requires the development and implemention of more innovative solutions. The spent fuels are no longer the same. Highly enriched fuels, high burnup fuels (over 60 000 MW·d/t HM) with high heat load and high radiation levels are beginning to be used. The management of spent MOX fuel also presents challenges. To optimize the competitiveness of nuclear power it is necessary to anticipate these needs.

2. EVOLUTIONARY PACKAGING FOR EVOLUTIONARY FUELS: ANTICIPATING NEEDS

There is a consensus that nuclear power is undergoing a renaissance. With the coming increase in electricity generating capacity from nuclear power plants utilities will have increasing amounts of spent fuel to manage, which will add to the existing amounts currently stored in reactor cooling pools. The first challenge is to store or transport these increased amounts of spent fuels.

Fuel specifications have been modified for evolutionary Generation III nuclear power plants, which will have increased burnup and fissile radionuclide content. These spent fuels will require advanced storage and transport container designs with enhanced neutron absorption capabilities, improved shielding, better thermal performance, higher structural resistance and increased containment to maintain the appropriate level of safety.

Design and safety principles must be consistent with the principles guiding the evolution of all fuel cycle facilities and with the two back end options — interim storage or reprocessing.

Dry metallic casks for interim storage or transport casks for reprocessing must first and foremost be designed to meet type B(U) requirements of the IAEA transport regulations [1], which is the international safety standard for the transport of radioactive materials. In the case of interim storage, a special safety analysis is required, including proof of the long term satisfactory behaviour of the materials.

Safety and environmental considerations in relation to spent fuel casks are also important issues, affecting not only the choice of packaging design technology, but also the capacities of storage pads and other constraints, such as the equipment required to manage the casks.

3. OPERATIONAL FLEXIBILITY AND THE DEVELOPMENT OF COST-EFFECTIVE SOLUTIONS

In relation to spent fuel storage, utilities in many countries are looking for comprehensive service coupled with a high level of safety. To provide this, cask designers must combine proven technical solutions and operational flexibility, along with specific solutions for each type of fuel and fuel history. It is important to provide operators of nuclear facilities operators with the latest technical developments to facilitate the loading and shipment of their fuels, to achieve a satisfactory reduction of radiation doses and costs, and to protect the environment. Moreover, research into higher cask performance must anticipate the need to quickly move and transfer fuel assemblies unloaded from a reactor core, which involves higher thermal and radiation specifications. Finally, new packaging systems should offer simplified maintenance, increased component life, as well as lower life cycle costs for transport equipment.

4. RESEARCH AND DEVELOPMENT OBJECTIVES

In order to address the needs of transport or interim storage of spent fuels from current and Generation III nuclear power plants a consistent research and development programme must be implemented, taking into account anticipated evolutions. The objective of TN International's research and development programme is to anticipate the needs described above in an increasingly demanding regulatory environment.

The programme focuses on four key areas:

(1) Improved basket technologies and materials;
(2) Innovations for heat transfer;
(3) New neutron shielding materials;
(4) Enhanced performance of shock absorbing covers (SACs).

4.1. Improved basket technologies and materials

One of the most important issues is the development of high performance spent fuel package design solutions to guarantee subcriticality. The general trend towards high burnups for LWR fuels (typically 60 000 MW·d/MTU for the EPR) leads to higher fissile contents — either increased U-235 enrichments (5%) or higher plutonium contents for MOX (9%). Subcriticality is guaranteed by the basket geometry and the neutron absorbing material.

Compactness is one of the major criteria for evolutionary cask design, as mass and volume are generally limited due to regulatory requirements or interfaces (e.g. the maximum allowable mass that can be transported or lifted by cranes). To achieve a high performance design, the choice of neutron absorbing material and innovative basket assembling methods are essential.

Boron (mainly the ^{10}B isotope) has been shown to be the most efficient neutron absorbing material, but it must be mixed in a metal alloy. The properties of borated alloys used in baskets must combine a high boron content for efficient neutron absorption, homogeneity, corrosion resistance and mechanical properties, even at 300ºC; and all at a reasonable price.

Research has been carried out on borated stainless steel plates (ASTM-A887), cast aluminium alloys, extruded aluminium profiles containing TiB_2, and metal matrix composites (MMCs) formed by casting, powder metal

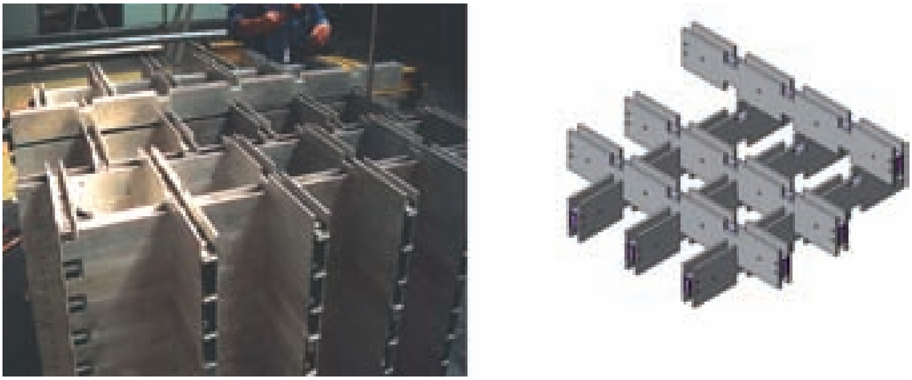

FIG. 1. Basket assembling.

processes or thermal spray. The high level of boron in MMCs allows the use of natural boron instead of enriched boron. The final products are formed by casting, rolling or extrusion.

An example of a high performance material for subcriticality is Boralyn™ (US patent 5700692) with 15% B_4C and natural boron. It can be used to ensure the structural resistance of baskets: yield stress at 280ºC is 280 MPa. Another example is the MMC developed with ALCAN: an aluminium matrix that will be used for TN24E storage packagings in Germany. It can contain up to 25% boron, and the product shows excellent homogeneity and high structural resistance without being brittle. These characteristics are very important for safety analysis in accident conditions.

Examples of innovations in basket assembling methods are shown in Figs 1–4.

Baskets made of stacked aluminium H profiles are shown in Figs 1(a) and (b) and 2(a) and (b) (Patents US 5881120 and International Patent application WO 2006/005891). Special assembling methods have been developed for basket walls with dovetail assembly (Fig. 3) (patent application EP 1378917) or nuts and bolts (Fig. 4) (patent application EP 1580763).

4.2. Innovations for heat transfer

Increases in burnup augment the residual heat of the fuel assembly. The first action to anticipate higher heat loads has been to improve the heat transfer capacity of packagings. Aluminium alloys are used for baskets because of their high thermal conductivity, but also because of the high emissivity of the surfaces (when anodized), inducing higher efficiency of thermal exchanges by radiation.

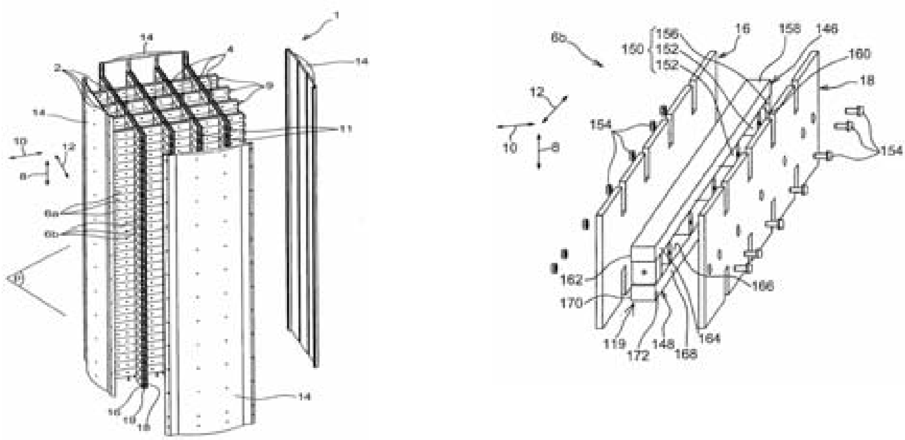

FIG. 2. Basket assembling: aluminium H profiles.

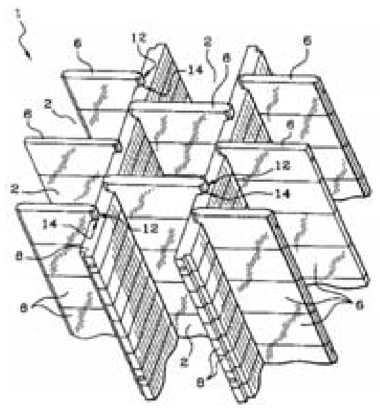

FIG. 3. Basket assembling: dovetail.

New heat dissipation systems are being considered for packaging. For heat transfer by convection, the optimization path includes an innovative fin design (Fig. 5 shows a steel cask external surface weld assembly with copper fins, patent US 6187395), or new helicoidal fins (Fig. 6), the use of helium in the packaging cavity, and heat transfer improvement of transfer canopies. Research is being carried out on adapting existing solutions to develop high performance casks.

The second action to anticipate increased heat loads is to deal with the effect of higher temperatures on components and materials. The new induced thermal profiles of casks requires additional analysis and, in some cases, new

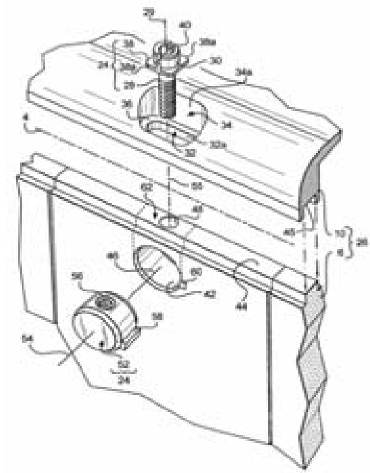

FIG. 4. Basket assembling: nuts and bolts.

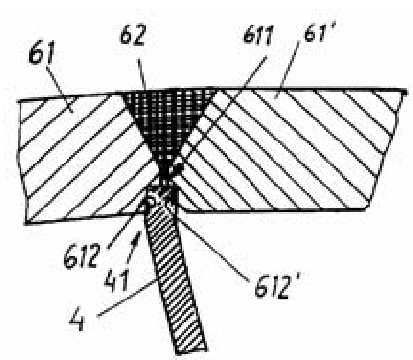

FIG. 5. Copper fin assembling.

materials are required that better resist higher temperatures. New materials are being studied and developed with this goal in mind, and a catalogue of temperature-resistant neutron shielding resins has been created. These resins are important tools for cask designers.

Concerning thermal behaviour, the new neutron shielding TN™ Vyal (patent application WO 03/050822) has the attribute of very high long term temperature resistance characteristics, higher than the thermoset polyester resins that are currently used. Ageing tests have shown that this material is effective at 160ºC [2], and work is under way on a new family of neutron shielding materials with even better properties (the target is 180ºC).

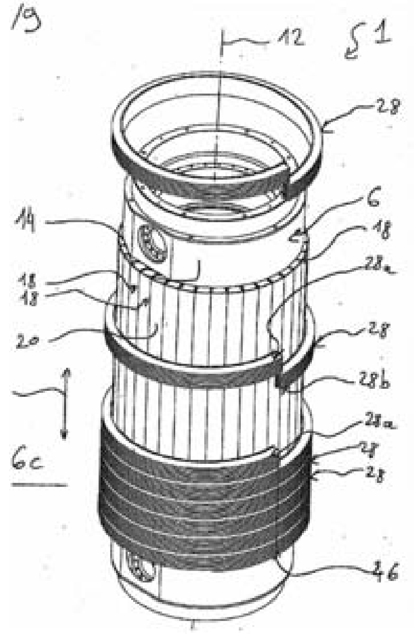

FIG. 6. Helicoidal fins.

A new generation of elastomer gaskets has been developed, called TN™ — Flex [3]. These gaskets resist a wide range of temperatures, from –40ºC to 200ºC in normal conditions, and up to 230ºC in accident conditions.

The behaviour of special aluminium MMCs for baskets has also been studied. The structural resistance of baskets is important under accident conditions; in these circumstances packaging must provide satisfactory characteristics, such as toughness or elongation at rupture, over the entire range of specified temperatures. This has been shown to be achieved with high boron content MMCs.

4.3. New neutron shielding materials

Fuel burnup optimization involves high initial enrichments, and the resulting spent fuels will present higher gamma and neutron radiation levels than current fuels. In the case of a quick evacuation of fuel from a reactor, the levels would be particularly high. Designers must anticipate these enhanced levels, especially in relation to evolutionary reactors. It is necessary to comply with regulatory requirements on site and during transport and, therefore, to work towards achieving the 'as low as readily achievable' criterion in relation to the control of radiation doses to workers and the public. With this in mind, efforts are being focused on developing enhanced shielding designs.

From Table 1 it can be seen that TN™ Vyal resin neutron shielding offers a higher shielding capability with a higher hydrogen and boron content than previous resins. The hydrogen content is 5.1×10^{22} atoms per cm^3 on average (not minimum), and the boron content averages 8.7×10^{20} atoms per cm^3. TN™ Vyal resin has been qualified in Germany for use in storage casks for vitrified waste and spent fuel. The product has undergone appropriate stability and quality tests (see Fig. 7).

The Vyal resin has excellent fire resistance; it is self-extinguishable and is classified M1 according to the NF-P92-501 standard and F0 according to the NF-F 16-101 standard, which covers the toxicity of smoke gases. From an environmental point of view, the main filler component of this resin, aluminium

TABLE 1. CHARACTERISTICS OF SOME NEUTRON SHIELDING MATERIALS

TN neutron shielding materials	TN12	F	VYAL	HYDROXANE	BORA
H 10^{22}at/cm^3	4.3	5	5.1	5.7	4.1
B 10^{20}at/cm^3	9	9	8.7	9	99
Density	1.45	1.8	1.8	1.4	1.76

FIG. 7. Neutron shielding material: Vyal resin qualification test.

hydrate, is a flame retardant that is not hazardous and not harmful to the environment [4]. In long term use at 160ºC, shielding performance is maintained as the Vyal resin remains stable and shows no visual or chemical degradation. Finally, the maximum allowable temperature of TN^{TM} Vyal resin is higher than previous resins.

Another advantage of this product family based on thermoset polymers is that it can be poured in situ even in complex geometry, using a pouring machine if necessary. With this system, it is easy to adapt resin thickness and to limit cost. Further work is under way to increase heat resistance and to reduce the package mass of neutron shielding materials.

4.4. Enhanced performance of SACs

Specifications to cover transport accident conditions are based on national regulations governing the transport of radioactive material which, in turn, are based on IAEA recommendations [1]. Type B packaging must resist a 9 m drop test and a puncture test (1 m drop onto a punch bar). Subcriticality, containment of radioactive material and a radiation dose rate limit must be maintained after an accident. To achieve this, SACs are attached to each end of the packaging to limit accelerations to an acceptable level. It is desirable to produce containment vessels and basket geometries with a high resistance but a limited mass.

Acceleration levels in a drop test are directly linked to the materials and dimensions chosen for the shock absorbing structures. Shock absorbing covers have been developed and tested by TN International for many years, and on a variety of casks with different mass, geometry, and specified accelerations. Current designs use wooden SACs and the optimal arrangement of the pieces of wood inside the SAC has been determined, with the grain oriented to absorb impacts in all possible directions, for the best performance of the SAC.

Dynamic crushing (Fig. 8) at various temperatures, with specimens of various sizes, has allowed several types of wood and their qualities to be classified. Balsa, redwood, red cedar and many other types of wood have been studied; the studies have included measurements of crushing characteristics, crushing stress and crushing length before bottoming, and recording variations according to different parameters (grain direction, temperature, speed, moisture, etc.). These data allow the worst impact cases to be evaluated. Tests have been performed on small specimens, with diameters of 40 mm in quasi-static conditions and on scale models at one third, half scale and even full scale in dynamic conditions. The results of these tests allow the behaviour of the SAC during impact to be predicted.

FIG. 8. Dynamic testing of wood.

Calculations with the computer code LS Dyna 3D are currently being used to predict accelerations, deformations and stresses. The effect of the steel structures of the SAC (gussets) is well known, as is the relation between the static and dynamic characteristics of wood. All this experience has been obtained through intensive testing with the assistance of and verification by experts of the competent French authority.

New technologies and materials for innovative SACs have been studied, including solutions other than wood. Wood has some disadvantages; it is not an isotropic material, and as crushing properties depend upon grain direction, wood pieces in the SAC must be arranged correctly. Foam materials can be isotropic or quasi-isotropic, as can hollow spheres made of metal or other materials. Phenolic foams and polyurethane foams are used in several existing packagings, such as the Trupact II package designed by Pactec USA.

Aluminium disks or rings offer high shock absorbing properties and their high crushing stress allows their use for heavy casks. Use of this type of material anticipates the need for better specific energy levels (impact energy that can be absorbed divided by mass of the shock absorber). For example, the TN™ 97L dual purpose spent fuel storage and transport cask (Fig. 9) has been licensed with an aluminium SAC.

Aluminium is not the only metal used for shock absorption. The properties of stainless steel have been tested in relation to plastic deformation without rupture for packaging, complying with type C regulatory requirements (air transport of radioactive material), which include a high speed impact corresponding to an aircraft crash.

5. CONCLUSION

With the increase in burnup, plant life extension and the need for higher flexibility in emptying reactor pools, a major research and development programme for spent fuel storage casks has been undertaken in anticipation of

FIG. 9. Aluminium SAC TN™ 97 packaging.

expected improvements in fuel design and spent fuel management. New packaging is being designed to meet these needs. For higher fissile contents, solutions have been developed with new neutron absorbing materials to ensure subcriticality. For higher neutron sources, new high performance shielding materials have been developed. And to allow for higher residual heat, new thermal designs have been developed.

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DEMONSTRATION TEST PROGRAMME USING FULL SCALE METAL AND CONCRETE CASKS

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Abstract

As a contribution to the development of safety standards for concrete casks, the Central Research Institute of Electric Power Industry of Japan has conducted a four year testing programme on concrete cask storage technology. A new research programme on the verification testing of cask integrity under long term dry storage conditions has been started. The paper summarizes these research programmes and the results obtained from them.

1. INTRODUCTION

According to the Framework for Nuclear Energy Policy [1] issued by the Atomic Energy Commission in October 2005, spent fuel will be reprocessed within the available reprocessing capacity and the surplus spent fuel will be placed in intermediate storage. Studies on the measures to be taken for the spent fuel stored at such interim storage facilities and spent MOX fuel from light water reactors will start around 2010. The construction of interim storage facilities for spent fuel at NPP sites or away from NPP sites, is expected to start in the near future.

In April 2006, the Nuclear and Industrial Safety Agency (NISA) issued technical requirements for interim spent fuel storage facilities (ISFs) using dry metal casks and concrete casks [2, 3]. In Japan, the first ISF away from a NPP site is being planned. Its commercial operation is expected to begin in 2010 in Mutsu City, Aomori prefecture (Fig. 1.).

In parallel with the regulatory and promotional activities on ISF, the Central Research Institute of Electric Power Industry (CRIEPI) has been performing supporting research studies related to the regulation and early implementation of ISF. Key aspects of these studies include the safety requirements for operation of spent fuel storage facilities, for unloading and loading during transport, and for the long term integrity of metal canisters and concrete materials, etc. CRIEPI has already completed a research programme on

FIG. 1. First interim storage facility away from an NPP site in Japan.

demonstration testing for the interim storage of spent fuel, mainly involving concrete cask storage technologies. These were carried out in support of the technical requirements being developed and now issued by NISA. A new research programme on the verification testing of cask integrity under long term dry storage conditions has now been started. The schedule of these programmes is shown in Table 1. This paper summarizes the research programmes and the results obtained from them.

2. CONCRETE CASK PERFORMANCE TEST WITH FULL SCALE CASK

2.1. Fabrication of the full scale concrete cask

In order to perform the heat removal tests and drop tests at full scale, a demonstration test facility was constructed in the Akagi Test Center of CRIEPI, located in the north, about 130 km from the centre of Tokyo (Fig. 2.). In the drop test, a steel plate is fixed to the base concrete. The steel plate is 7.5 m long, 4.5 m wide and 50 mm thick; the thickness and weight of the base concrete are 2 m and 550 tonnes, respectively. Full scale concrete cask performance tests, including heat removal tests, drop tests and seismic tests were completed during the phase 1 and phase 2 programmes.

Strength and safety must be maintained under the conditions in which the casks are used, some of which are peculiar to Japan. The preliminary design parameters are shown in Table 2. The concrete cask is assumed to be for use within a storage building. Two types of concrete cask to store the high burnup

TABLE 1. SCHEDULE OF DEMONSTRATION PROGRAMMES FOR STORAGE CASKS

Programme item	2000	2001	2002	2003	2004	2005–2008
Concrete cask performance test			Phase1			Phase2
(a) Basic design						
(b) Fabrication of full scale concrete cask						
(c) Demonstration tests heat removal						
MPC drop						
Seismic	1/3 scale					Full scale
(d) Safety analysis						
Containment performance test of metal cask			Phase1			Phase2
(a) Drop test without impact limiter						
(b) Long term sealability test of lid structure						

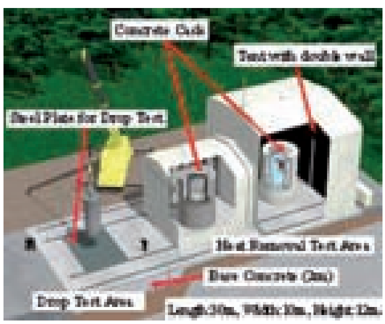


FIG. 2. The demonstration test facility at the Akagi test centre of CRIEPI.

spent fuel were designed, a reinforced concrete cask (RC cask) and a concrete filled steel cask (CFS cask) (Fig. 3, Table 2).

The RC cask is made from reinforced concrete which is a component of the cask's structural strength. In the CFS cask, the storage container consists of concrete covered with a steel sheet, creating a steel structure. In this case, the

TABLE 2. PRELIMINARY DESIGN PARAMETERS

Design storage period	40~60 a
Fuel type	17 × 7 array for PWR
Enrichment (wt % U^{235})	4.9%
Burnup (Max)	55 MW·d/kg HM
Cooling time	10 a
Environmental temperature	33°C
Storage cell	21
Total heat load (max)	22.6 kW

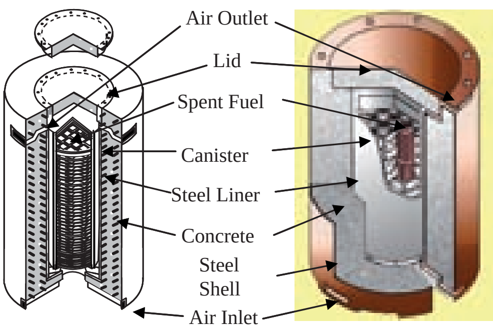


FIG. 3. Outlines of the two types of cask.

concrete is not a component of the structural strength of the cask but, rather, a radiation shielding material. The two types of the concrete cask were constructed in the test facility as shown in Fig. 4.

Two types of full scale MPC were designed and fabricated, as shown in Fig. 5. Each canister has a capacity of 21 PWR spent fuel assemblies. The canister body is made of highly corrosion resistant material. Basket of type I consists of guide tubes and stainless steel plates. The stainless steel plates are fixed at constant intervals by steel rods and each has 21 square holes for the guide tubes. The guide tubes are placed in the holes and fixed to the plates. To increase thermal conduction, an aluminium plate is fixed to the stainless steel plate. The type II basket is an assembly of rectangular hollow blocks made of aluminium alloy. During the welding procedures, a helium leak test,

FIG. 4. Appearance of the full scale casks.

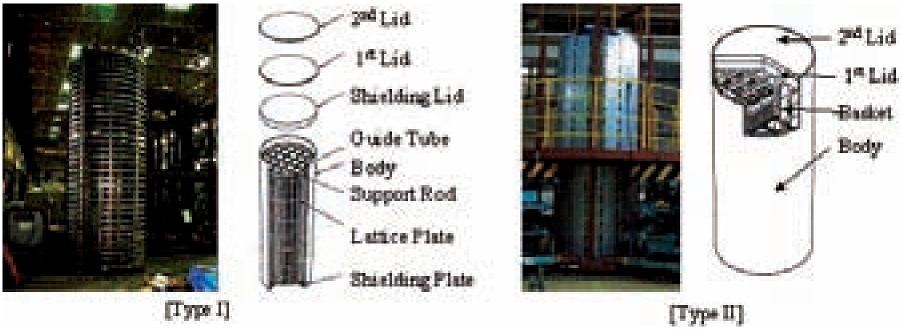


FIG. 5. The two types of test MPC.

penetration tests and ultrasonic tests were carried out to ensure the quality of the welding.

2.2. Heat removal test

To measure the temperatures at each position in the cask and the flow rate under normal conditions and in accident conditions, and to evaluate the integrity of the components of the cask, heat removal tests on the two types of full scale casks described above were carried out [4, 5].

TABLE 3. SPECIFICATIONS OF THE CONCRETE CASKS

Cask type	RC	CFS
Height	5787 mm	6120 mm
Outside diameter	3940 mm	3800 mm
Inside diameter	1850 mm	1838 mm
Total Weight	185 t	184 t
Canister type	Type I	Type II
Height	4630 mm	4470 mm
Outside diameter	1676 mm	1640 mm
Weight	35 t	30 t
Body material	Super stainless steel	Austenitic-ferritic stainless steel
Basket material	Stainless steel	Aluminium alloy

2.2.1. Normal condition

The decay heat of spent fuel was simulated using electric heaters with the heat rate of the initial storage stage, corresponding to 22.6 kW, and of the final storage stage, corresponding to 10 kW.

The RC cask

Figure 6 shows the temperature distribution on the concrete inside of the cask for the test of the initial storage stage with heating of 22.6 kW. The maximum value in the vicinity of the outlet duct was 91°C, and it exceeded the allowable temperature limit value for the long term; i.e. 90°C. It was considered that the pressure loss was large in the flow channel of the cask and, therefore, that the thermal design, for example, the flow channel shape, should be changed.

Figure 7 shows the heat balance for the removal of the decay heat from the cask. It was found that 80% of the total heat was removed by the cooling air and that the heat transfer from the bottom of the concrete container to the floor was insignificant. These data may be useful for the design of the cask.

The CFS cask

At the initial storage stage, with heating of 22.6 kW, the maximum temperature of the concrete container was below the allowable temperature limit for the long term period.

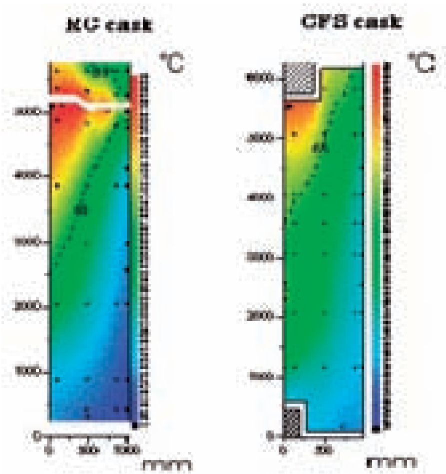


FIG. 6. Temperature distributions of the inside concrete under normal conditions.

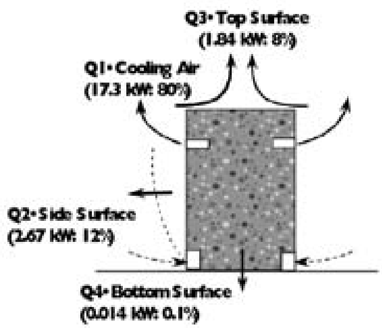
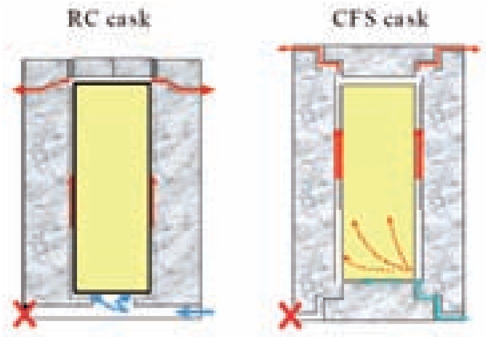


FIG. 7. Heat balance of the RC cask.

In both casks, the canister surface temperature was below 100°C in the final storage stage with 10.0 kW heating.

2.2.2. Accident conditions

To simulate accident conditions, tests were conducted with two of the four inlets blocked. In the RC cask, the air is uniformly distributed to the annulus gap from the bottom cavity. In the CFS cask, the air flows to the annulus gap directly, as shown in Fig. 8.



'X' indicatess the position of inlet blockage.

FIG. 8. Flow patterns with a 50% blockage of the inlet ducts.

In both casks, the maximum temperatures in the concrete container were under the allowable temperature limit value for the short term, i.e. 175°C, within 24 hours, as shown in Table 4. The temperature values in this table are normalized at the inlet air temperature of 33°C. The increase of temperature in the concrete container was 5°C in the RC cask and 10°C in the CFS cask. The decrease in the rate of air flow was 4% in the RC cask and 23% in the CFS cask. The flow pattern is clearly different between the RC cask and the CFS cask. However, it was found that both casks had sufficient heat removal capacity under accident conditions.

2.3. Drop test [7]

Table 5 shows the drop test conditions for the full scale canisters. Two drop tests in horizontal and vertical orientations were conducted to simulate drop or impact events during handling operations with drop heights of 1 m and 6 m, respectively. Regarding the contents of the canister, dummy steel structures equal to the total weights of the spent fuel (14.7 t) were used.

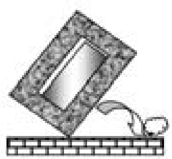
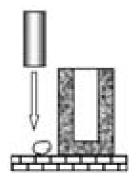
During the horizontal drop test, the test canister was slightly deformed near the impact area. The time histories of accelerometers and strain gauge readings at various points in the test canister were recorded. The average deceleration value was about 436 g at the top of the lids. On the other hand, in case of the vertical drop test, although the bottom plate of the test canister was deformed by the force of inertia of its contents, the deformation of the bottom of the basket was negligible. The average deceleration value was about 1153 g at the centre of the shell.

TABLE 4. TEMPERATURE AND FLOW RATE AT 22.6 KW UNDER NORMAL CONDITIONS AND IN AN ACCIDENT CONDITION OF 50% BLOCKAGE OF THE INLET

Cask type / Item	RC cask		CFS cask	
	Normal	50% blockage	Normal	50% blockage
T_{in} of air (°C)	33		33	
T_{max} of concrete body (°C)	91 (90[1])	96 (175[2])	83 (90[1])	93 (175)
T_{max} of canister surface (°C)	209	214	192	200
T_{max} of guide tube (°C)	301	306	228	235
fT of air (°C)	65	70	52	66
Flow rate (kg/s)	0.335	0.321	0.363	0.280

[1] Allowable temperature limit value for the long term: 90°C [6].
[2] Allowable temperature limit value for the short term: 175°C (within 24 hours).

TABLE 5. DROP TEST CONDITIONS

Canister	Type I	Type II
Non-mechanical drop or impact events during handling	Tipping over Event 	Drop Event 
Orientation	Horizontal 	Vertical 
Height	1 m[1]	6 m[2]

[1] Equivalent drop height for rotational velocity caused by tipping over from height of GC.
[2] Drop height from cask height.

FIG. 9. Helium leak test for the canister.

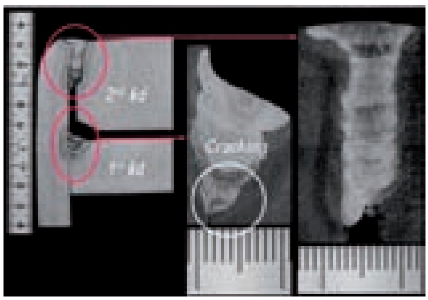


FIG. 10. Magnified view of the cut section of the directly impacted welded part of the horizontal drop tested canister.

Helium leak tests, as shown in Fig. 9, were performed before and after the horizontal and vertical drop tests to confirm the integrity of leaktightness of the test canisters (especially welded lids) against impact loads. Measured leakage rates in both tests showed that the integrity of the lid and canister shell seals had been maintained, that is all values were under 1.0×0^{-9} Pa.m^3/s. Figure 10 shows a microscopic view of the section of the directly impacted welded part during the horizontal drop test (magnified 5.7 times). Crack initiation was found — possibly due to the impulsive moment around the top corner of the test canister. However, the initiated crack was arrested in the first welded layer. In the case of the vertical drop test no crack initiation was found, although a small air blow hole was detected.

2.4. Seismic test

The concrete cask is intended to be oriented vertically in the freestanding condition. To evaluate the tipping over phenomena under a strong earthquake

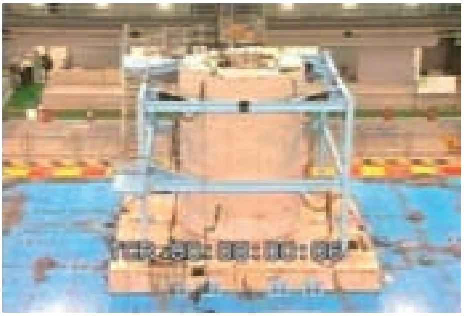


FIG. 11. 3-D full scale earthquake testing with full scale concrete cask and concrete floor.

motion, excitation tests were performed on a one third scale model concrete cask using a two dimensional shaking table, and the applicability of the energy spectrum approach for the estimation method of the tipping over event was verified [8]. However, as the concrete cask has multiple gap structures, such as the annulus space for cooling air between the canister and concrete container and the gap between the spent fuel and basket, the seismic response of the spent fuel is complicated.

2.4.1. Test condition

The full scale concrete cask and storage house floor were set on a three dimensional shaking table in a 3-D full scale earthquake testing facility ‘E-Defense’, designed and constructed by the National Research Institute for Earth Science and Disaster Prevention (NIED), as shown in Fig. 11.

One full scale PWR fuel assembly (17 × 7) and 20 dummy PWR fuel structures, for which the model deformation was equivalent to the full scale deformation, were fabricated. An 80 cm thick and 8 m wide reinforced concrete slab (weight 125 t) was used as the storage house floor model. During the seismic excitation test the angle, angular velocity, acceleration and displacement of the cask body, canister and fuel structures were measured, as shown in Fig. 12. During the seismic excitation test, recorded waves from typical natural earthquakes and artificial seismic waves were employed. Test conditions included cases with simultaneous horizontal and vertical motions.

2.4.2. Test results

During the seismic response of the cask three dimensional behaviour including top spinning and sliding movements were observed. However, the

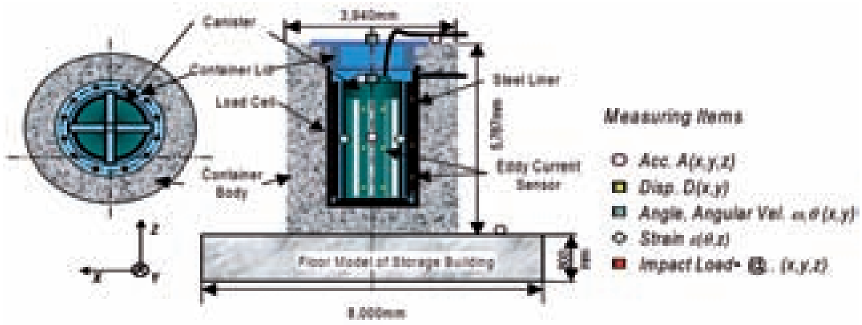


FIG. 12. Full scale concrete cask for seismic test.

full scale cask did not tip over, even when the acceleration level exceeded the ultimate input level for tipping over derived by the energy spectrum method or by the tipping over velocity criteria approach. Moreover, since the deformation of the spent fuel structures still remained in the elastic region under the strong seismic motion, it was concluded that the integrity of the spent fuel is maintained under Japanese seismic design conditions.

3. CONTAINMENT PERFORMANCE TEST OF THE METAL CASK

3.1. Drop test without impact limiters under accident conditions

In an interim storage facility, metal casks will be handled without impact limiters. Although there have been many reported tests and analyses on the evaluation of metal cask drop tests, no quantitative measurement has been made of instantaneous leakage through metal gaskets subjected to impulsive loads. In this study, leak tests were performed using a full scale metal cask without impact limiters in drop accidents during handling in a storage facility [9]. The instantaneous leak rate was quantitatively measured during the drop tests.

3.1.1. Test condition

Figure 13 shows the conditions of the drop test. A series of impact tests were carried out (a horizontal drop test from a 1 m height and a rotational impact test around an axis of a lower trunnion of the cask from the horizontal orientation at a 1 m height) on to a reinforced concrete slab simulating the floor structure of the facility. Measurements were made of the changes in the

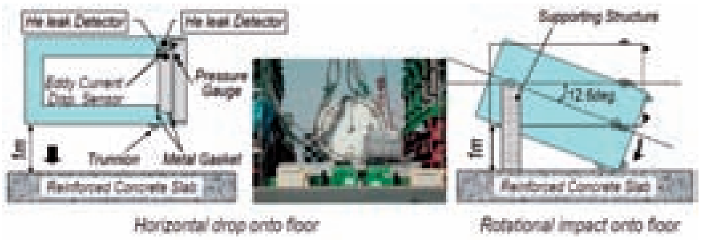


FIG. 13. Drop test conditions for the full scale metal cask.

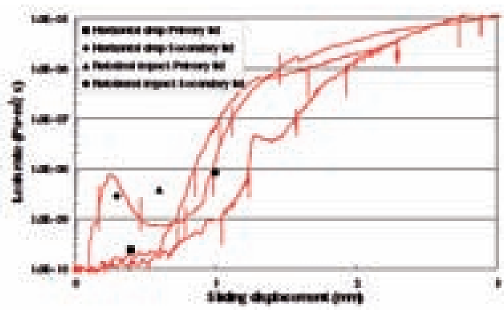


FIG. 14. Relationship between leak rate and sliding displacement.

sliding and opening displacements of the primary and secondary lids, of the leak rates, and of the pressure in the volume between the two lids.

3.1.2. Test results

Figure 14 shows the time histories of the leak rate from the secondary lid during a rotational impact test. Although the leak rate value from the secondary lid increased by two orders of magnitude at the moment of impact, the leak rate recovered to the background level value within 20 minutes of the drop test. The change in the pressure between the two lids was negligible.

Figure 15 shows the relationship between the maximum sliding displacements of the lids and the leak rates obtained with the static loading tests and as predicted by the scale models. The relationship between the maximum sliding displacement of the lids and the maximum leak rate of the full scale metal cask subjected to impulsive loads showed good agreement with that predicted by the scale model gasket structure.

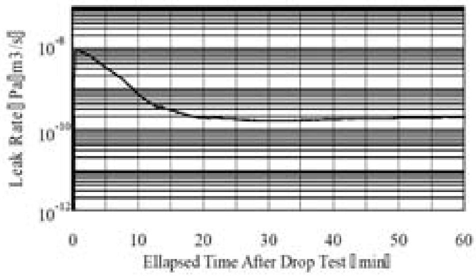


FIG. 15. Time histories of the leak rate from the secondary lid (rotational drop test).

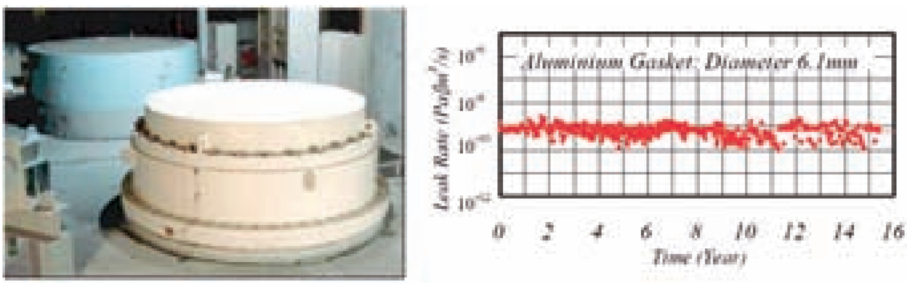


FIG. 16. Sealing tests with two full scale lid models.

3.2. Containment performance of metal cask during long term storage

The confinement structure of the metal cask is designed to be highly reliable as a result of the use of metallic gaskets instead of conventional rubber gaskets. It is very important to clarify the influence of the stress relaxation of the gaskets on containment performance in the long term. The long term containment of the secondary metal gasket in the cask lid structures of the full scale models has been measured for more than 15 years at a constant temperature of 140°C, as shown in Fig. 16. The results indicate that the containment will be maintained for more than 50 years, taking account of the decay heat of the spent nuclear fuel.

3.3. Future key research issues

In Japan, the dry metal cask has been receiving the highest priority for implementing storage facilities in the short to medium term, mainly because of its superior economics compared to water pool storage. With the longer term

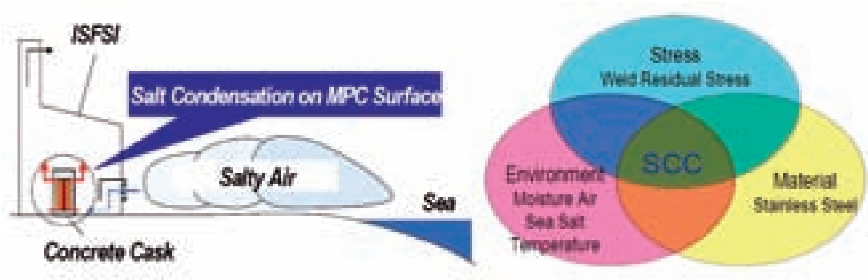


FIG. 17. Deterioration of metal canisters in a salt water environment.

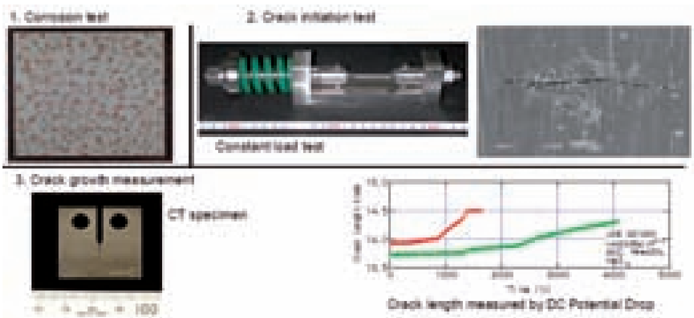


FIG. 18. SCC evaluation tests.

perspective in mind, research on the utilization of the dry dual purpose metal canister is now under way for other dry storage technologies, aiming at better economic performance.

A key issue for the implementation of metal canister storage technology is the long term integrity of canisters, with special consideration being given to the deterioration of metal canisters in a salt water environment as interim storage facilities in Japan are likely to be installed at coastal sites. Temperature will decrease during the storage period and salt condensation will increase on the metal canister surface. As austenitic stainless steel may be vulnerable for SCC under certain circumstances, as shown in Fig. 17, it is important to prevent penetration through the wall by SCC. From 2004, CRIEPI started SCC evaluation tests as shown in Fig. 18 to clarify basic deterioration mechanisms as follows:

(a) Finding the threshold value of chloride density to initiate SCC for the stainless steel canister surface;

(b) Estimation of the chloride accumulation transported by cooling air on the metal canister surface;
(c) Lifetime of the canister material against SCC penetration.

4. CONCLUSIONS

In Japan, utilities are planning to commence operation of the first interim spent fuel storage facility in 2010. The regulatory authority has modified the reactor regulation law and has been establishing the relevant safety rules for the operation of ISFs. To help prepare the safety requirements and to develop a safety review procedure for the licensing of the interim storage facility, CRIEPI is steadily performing the key research studies, which include degradation of cask component materials, leakage from the lid during accidents during the subsequent transport after storage.

ACKNOWLEDGEMENTS

This work has been carried out under contract from NISA/METI.

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FUEL AND CLADDING PROPERTIES AND BEHAVIOUR

(Session 6)

Chairperson

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Germany

DAMAGE IN SPENT NUCLEAR FUEL DEFINED BY PROPERTIES AND REQUIREMENTS

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Abstract

The Nuclear Regulatory Commission's (NRC's) Spent Fuel Program Office (SFPO) has provided guidance in defining damaged fuel in Interim Staff Guidance, ISG-1. This guidance is similar to that developed by the American National Standards Institute (ANSI). Neither of these documents gives the logic behind its definition of damaged fuel. The paper discusses the requirements placed on spent fuel for dry interim storage and transport and the ways in which service requirements drive the definition of damage for spent fuel. Examples are given to illustrate the methodology, which focuses on defining damaged fuel based on the properties that the fuel must exhibit to meet the requirements of storage and/or transport.

1. INTRODUCTION

Both the Nuclear Regulatory Commission, via Interim Staff Guidance ISG-1 [1], and the American National Standards Institute (ANSI) [2] have provided definitions of damage for application to spent fuel. While these definitions are linked to regulations, explanations are not provided as to why the fuel, under these definitions, should be considered damaged. This paper provides explanations for the definitions of damaged spent fuel. In each case, the rationale is based on the ability of the fuel to meet a regulatory requirement for a particular phase of the nuclear fuel cycle. Due to the variety of regulatory requirements at various stages of the nuclear fuel cycle, the meaning of damage varies, depending on the requirements for each stage of the life of the fuel in the reactor or in the post-reactor period. Thus, the potential exists for there to be independent definitions for interim storage and for transport. During each stage in the life of the fuel, the definition of damage is related to the requirements of the fuel that are specific to that stage.

Fuel rods and assemblies undergo many changes from the time they are manufactured until they are removed from the reactor, and these changes can alter their mechanical properties. Numerous factors bring about the changes: radiation damage, cladding oxidation and thinning, creep-down of the cladding, and the presence of hydrides formed as byproducts of the oxidation process. The physical properties and characteristics of the rods are also affected by reactor service.

The effective thermal conductivity is decreased by a coating of 'crud' that forms on the rod surface during irradiation, creep-down, and by the release of the fission products and gas into the plenum region:

(a) The rods and assemblies elongate and the assemblies bow;
(b) The effective localized thickness of the cladding decreases by abrasion from debris floating in the reactor coolant and by vibrations that lead to rod fretting against the grid spacers;
(c) Inside the rods the fuel pellets crack into numerous pieces.

Any of these alterations of the mechanical and physical properties could be considered to represent damage to a fuel rod. Fuel rods are considered damaged only if they cannot function as required for post-reactor conditions and operations. Fuel that is considered damaged for any other part of the fuel cycle may or may not be considered damaged for post-reactor operations, as the expectations of the fuel performance may be different. The term 'damaged' can be defined by the ability of the fuel to perform its intended function in a given phase of the nuclear fuel cycle.

2. NRC REGULATIONS FOR SPENT FUEL STORAGE AND TRANSPORT

The US Federal Regulations [3, 4] contain requirements relating directly to the condition of spent fuel during storage and transport. The regulations for spent fuel storage[1] stipulate that the fuel must be retrievable, protected against gross ruptures and compatible with the rest of the system. While protection against gross ruptures is not defined in the regulations, the objective might be related to the prevention of the escape of fuel fragments from the rod into the

[1] 10CFR72.122 [l] addresses spent fuel retrievability, 10CFR72.122(h)(1) addresses gross ruptures of spent fuel cladding, and 10CFR72.236(h) addresses compatibility.

cask and to assurance that retrievability in the form of a rod is maintained. The regulations for spent fuel transport[2] require that the spent fuel is not substantially rearranged during normal conditions of transport. When the properties of spent fuel rods and assemblies do not meet the properties required by the approved conditions for storage and transport, the regulatory performance requirements are not met by the fuel and the fuel should be classified as damaged.

In addition to the regulations that specifically address the state of the spent fuel during storage and transport there are a number of regulatory requirements pertaining to criticality, shielding, thermal, containment or confinement, structural, and materials issues that may indirectly impose performance requirements on the fuel rods and fuel assembly. When an applicant requests NRC approval for a cask design, the applicant specifies the system, the materials to be used, the range and condition of the fuel to be stored (type, burnup, cooling time, etc.), and the conditions of storage such as temperature, atmosphere, and length of storage. The system is analyzed to ensure and demonstrate that all pertinent regulations are met. When this is done, any fuel assembly or fuel rod in the specified range that prevents the system from meeting these indirect regulations should be considered damaged.

3. EXISTING GUIDANCE ON THE DEFINITION OF DAMAGED FUEL

An applicant is required to meet the storage and transport regulations given in 10 CFR Parts 71 and 72 [3, 4] and is free to define damaged fuel in any manner, provided that the definition allows the regulations to be met. The NRC Spent Fuel Project Office (SFPO) staff has evaluated mechanisms that could affect the behaviour of spent fuel and spent fuel assemblies, and has proposed a definition for damaged fuel in interim staff guidance ISG-1. ANSI has developed a standard that defines damaged spent fuel with respect to storage and transport. Neither document gives a basis for the definition nor how the definitions satisfy the requirements of 10 CFR 71 and 72. Both documents provide guidance on other information such as records, quality assurance, and examination techniques useful for determining fuel rod condition. The pertinent parts, those related to the definition of damage, are

[2] 10CFR71.55(d) [2] addresses spent fuel configuration during normal conditions of transport.

compared in Table 1. They are very similar and appear to be related to the function of the fuel rods or assemblies.

TABLE 1. COMPARISON OF ISG-1 AND ANSI DEFINITIONS OF DAMAGED FUEL

	ISG-1 Rev 1	ANSI
Fuel rod breach	(1) The fuel contains known or suspected cladding defects greater than a pinhole leak or hairline crack that have the potential for release of significant amounts of fuel particles into the cask.	Cladding Damage, Level I. Cladding defects greater than pinholes or hairline cracks but the fuel assembly still remains intact as a fuel assembly
Debris	(2) The fuel is no longer in the form of an intact fuel bundle and consists of, or contains, debris such as loose fuel pellets, rod segments, etc.	Cladding Damage, Level II. Fuel that is no longer in the form of a fuel assembly and consists of debris, loose pellets and particles, rod segments, etc.
Structural	(3) The fuel assembly: (a) Is damaged in such a manner as to impair its structural integrity; (b) Has missing or displaced structural components such as grid spacers; (c) Is missing fuel pins which have not been replaced by dummy rods which displace a volume equal to or greater than the original fuel rod; (d) Cannot be handled using normal (i.e. crane and grapple) handling methods (4) The fuel assembly structural hardware or cladding material properties are in a degraded condition such that its ability to withstand the normal and design basis events of storage (for storage-only casks), or the normal and hypothetical accident	Fuel Assembly Damage. Fuel assemblies that have structural damage such that they cannot be handled by normal methods.

4. CONDITION OF FUEL, DEFINITIONS AND MECHANISMS OF DEGRADATION

4.1. Typical spent fuel

The condition of the SNF as it comes out of the reactor is the baseline for determining the behaviour of fuel in storage and transport. The typical condition of the fuel and the associated range of uncertainty is normally determined by poolside non-destructive and hot cell destructive examinations on representative numbers of assemblies, and by reviewing reactor records to determine when and how many cladding breaches have occurred. It is common practice in the industry to subject breached rods to a detailed examination. Usually the fuel rod has no through-cladding penetrations and the fission gas release from the pellets, cladding creep-down, cladding hydrogen impurities, oxide thickness and distortion of the assemblies due to bow, all fall within an established set of limits.

4.2. What is ‘a pinhole’, ‘a tight crack’, and ‘a gross breach’?

In early studies, the most prevalent breach mechanism was stress corrosion cracking (SCC). SCC caused cladding penetrations large enough to only release gases. With this size of breach the cladding can still satisfy the requirements to confine the fuel in the rods for operational safety and retrievability purposes. As a result, both ISG-1 and ANSI describe damaged fuel in terms of pinhole leaks and tight (hairline) cracks. Generally speaking, pinhole leaks and hairline cracks are breaches in the fuel cladding that do not allow the escape of fuel particulate material. The guidance in ISG-1 allows fuel to be classified as intact if only pinhole leaks and hairline cracks are present. From a retrievability perspective, a gross breach might be considered to be any cladding penetration that allows fuel to escape from the rod, i.e. any breach that compromises the confinement capability of the cladding. During irradiation, a pellet cracks into 10 to 30 pieces, excluding a small amount of ‘fines’ at the pellet–pellet interface. Although the fragments tend to be wedge shaped, a fractured, 10 g pellet could be approximated by 30, individual 3mm-diameter fragments, which would not be able to escape from a 1 mm breach. A tight (hairline) crack might be defined as any crack that does not visually expose fuel.

4.3. The fuel pellet oxidation process and cladding splitting

Irradiated uranium dioxide exposed to an oxidizing atmosphere will eventually oxidize to U_3O_8. The oxidation time is exponentially related to temperature, according to the Arrhenius Law. Initially the pellet grain boundaries are oxidized to U_4O_9, resulting in a slight matrix shrinkage and further opening of the structure [5]. The oxidation then proceeds into the grain until there is complete transformation of the grains to U_4O_9. A plateau in the process occurs at this time, until the fuel resumes oxidizing to the U_3O_8 state. The transformation to U_3O_8 occurs with a ~33% lattice expansion that tears the ceramic fragment structure into grain sized particles. When the UO_2 pellets are encased in cladding to form a fuel rod the swelling of the fuel pellets due to oxidation to U_3O_8 places a stress on the cladding. The cladding may experience strains of up to 6% before any initial defect starts to propagate axially along the rod [6].

During the oxidation process the fuel pellet fragments are reduced to a grain-sized powder that can easily escape from a damaged rod [7]. The extent of oxidation and cladding splitting in any particular rod will depend on the number of rods with cladding breaches, the free volume in the cask, and the temperature. The rate at which the cladding splitting occurs has been experimentally measured and modelled, but there are a number of variables that can affect the rate, such as fuel burnup, moisture content of the air, the cladding material and the type of initial defect. Depending on the temperature, over a 20 year storage period from 10 to 750 cm of the fuel column could oxidize and split the cladding [8].

4.4. Cladding hydride reorientation

From the time SNF is removed from the spent fuel pool until it reaches the repository it is subject to numerous mechanical forces (e.g. vibration) and a variety of thermal cycles. These forces and cycles can degrade the fuel and alter its ability to meet the requirements for criticality and retrievability. Of particular concern is the possibility of hydride reorientation during storage and subsequent reduction of the cladding mechanical strength during transport. During short term cask loading operations (including drying, backfilling with inert gas, and transfer of the cask to the storage pad), the fuel is subject to elevated temperatures and hydrogen goes into solution in the cladding up to the saturation concentration of the solvus. As the spent fuel cools the hydrogen re-precipitates. The orientation of the re-precipitated hydrides depends strongly on the hoop stress inside the fuel rod vis-à-vis the critical stress needed for reorientation.

Studies at the Argonne National Laboratory are currently under way to determine this critical stress for hydride reorientation as a function of cladding type, fuel cooling rate, temperature and other parameters. In addition, work is planned to determine the effects of the reorientation on the mechanical properties of the cladding as a function of cladding type, hydride concentration and temperature. Until these studies are complete the degree of cladding damage due to hydride reorientation cannot be completely assessed. Due to this uncertainty, NRC staff guidelines currently suggest that the cladding hoop stress be kept below the best available estimate for the value of σ_H^{cr}, which is 90 MPa at 400°C.

5. EXAMPLES[3]

The following are a number of examples in which the definition of damaged fuel rods or damaged assemblies is driven by either the regulations which specifically reference the condition of spent fuel (e.g. regulations addressing retrievability or gross breaches in storage) or those regulations that indirectly relate to the state of the spent fuel (e.g. regulations addressing criticality control). These examples demonstrate why it is important to consider function based definitions of damaged fuel.

5.1. Retrievability

5.1.1. Fuel damage

This example is concerned with the dependence of fuel classification on temperature and atmosphere; these are two of the parameters that may be controlled by the design of the storage system. Based on an extensive evaluation of potential degradation mechanisms for cladding in storage (e.g. creep and hydride reorientation), an upper fuel cladding temperature limit for storage of 400°C was recommended by the NRC staff in ISG-11, revision 3 [9]. At this temperature, an inert atmosphere must be maintained to prevent small breaches (pinhole breaches and hairline cracks) in the cladding from deteriorating into gross ruptures due to fuel oxidation. If an inert atmosphere is not used or the temperature is not sufficiently lowered, small breaches may split

[3] These examples are not to be taken as the NRC position or guidance but rather as illustrative of the concept of 'damage defined by function'.

open to form gross breaches; thus, spent fuel that was classified as intact[4] at the time of loading in the storage cask should have been classified as 'damaged' due to its potential to not meet the regulatory requirements while in storage. This would pose retrievability and ALARA (as low as reasonably achievable) concerns at the time of the removal of the spent fuel from the storage cask since gross breaches may lead to the formation of fuel debris that is not contained in a manner that allows for retrieval by normal means. This example demonstrates how the cask designer's informed choice of both the maximum allowable temperature and atmosphere in the storage cask can prevent fuel that is stored as intact from becoming damaged during the duration of storage. If an oxidizing atmosphere is used and the temperature is not appropriately controlled, fuel rods with any cladding breaches may have to be classified as damaged and stored as such, even if they do not meet the criteria of damaged fuel at the time of loading, to ensure that the fuel is retrievable later in the fuel cycle.

In addition to the environment of the storage cask itself it is important to give consideration to the environment present during cask loading operations. For example, if air is used to blow down loaded casks prior to lid welding and to completely void the cask of water it is possible that some of the uncovered rods or parts of rods might be exposed to air at elevated temperatures for extended periods of time. Under these circumstances, small breaches in rods defined as 'intact' could become gross ruptures which could then release fuel particulates to the cask interior. The regulatory requirement that the rods do not have gross breaches mandates that either the time–temperature history of the rods or the potential for rod breaches be considered in the definition of damaged fuel if an oxidizing atmosphere is used in cask blow-down operations.

5.1.2. Assembly damage

The assembly hardware is a vehicle for transferring the fuel. Fuel assemblies in storage, possibly excepting those in dual purpose (storage and transport) casks, will eventually have to be removed from the storage cask either to be placed into a transport cask or to be transferred to a disposal cask. In this case, for fuel to be classified as undamaged, all relevant past and current experience should indicate that a fuel assembly can be handled and moved using normal methods.

If a fuel assembly has been altered such that it may not be handled and moved by normal means then it does not fulfil its purpose and should be

[4] Based on the guidance in NRC ISG-1, revision 1 [1].

classified as damaged. Alterations to, or removal of, the fuel rods, grid spacers, grid straps, or other structural components or hardware may affect the way a fuel assembly has to be handled.

A fuel assembly, otherwise classified as damaged, may be analyzed against storage or transport requirements to determine if it may be considered as undamaged intact fuel. For example, an assembly that has part of a grid strap missing may be classified as undamaged if analysis shows that the damaged strap does not hinder the meeting of all of the transport requirements. This approach could avoid the requirement for using damaged fuel cans in some cases. Such analyses would have to demonstrate with reasonable assurance that the assembly can withstand the conditions of storage or transport and still meet retrievability requirements.

Assemblies with modified or repaired top bails, etc. may be classified as undamaged from a retrievability perspective since they readily permit the transfer of fuel. However, before classifying such an assembly the repair method must be evaluated. The evaluation must reasonably demonstrate that the repair will not degrade after either exposure to the high temperatures of dry storage or transport (relative to spent fuel pool temperatures) or if it is subject to other design conditions such as the hypothetical drop accident of transport. Fuel assemblies that are properly reconstituted and complete and that contain undamaged components of original type or equivalent, and are of original geometry, should not be considered damaged since they fulfil their intended retrievability function.

5.2. Radiation dose rate, containment and criticality control in transport

5.2.1. Fuel damage

The transport regulations in 10 CFR 71.51 [4] limit radioactive releases from a package and limit radiation dose rates under normal and accident conditions. The configuration of the fuel and the ability of the cladding to retain fuel particulate and fines are parameters that are used when assessing compliance with these requirements. The permitted degree of fuel damage is therefore a design basis assumption used for demonstrating that a package design meets the performance requirements of Part 71. It is required in 10 CFR 71.55 that during transport the fuel does not become rearranged under normal or hypothetical regulatory accident conditions, so that if a moderator floods the package a criticality event will not occur. Any fuel assemblies that do not meet these specifications may need to be classified as damaged.

5.2.2. Assembly damage

The fuel must not assume a configuration, during either normal conditions or postulated accident conditions, such that criticality could occur. In the structural analysis of the assemblies during an event it is usually assumed that specified components of the assembly such as grids, flow mixers, tie rods, etc. are in place and that the components have the properties associated with the given material, material state, irradiation level and transport temperature.

If a fuel assembly has certain components missing, has damaged components, or has been modified, it might still be classified as intact if it can be shown that the assembly in the defined configuration still meets the regulatory requirements. Should the structural integrity of the assembly be adversely affected under the design basis storage and/or transport conditions, then the assembly might not fulfil the requirement that it maintains its configuration and it should be considered damaged. The cask designer has the freedom to design a system that mitigates the forces transmitted to the assembly and fuel rods. If the storage cask or transport package design prevents or mitigates forces transmitted to its contents such that structural integrity is not significantly compromised, the assemblies need not be classified as damaged, assuming other factors (temperature, inert atmosphere, etc.) have been adequately addressed. This example illustrates how the design of the system can change the requirements that define an assembly as either damaged or intact.

5.3. Stress driven damage

If the hoop stress on the cladding exceeds the stress threshold due to in-reactor temperature excursions or crud buildup, for example over a large number of rods or length of rod, then hydride reorientation might occur. Should it be determined that hydride reorientation has degraded the properties of the high burnup SNF to the extent that it cannot maintain an acceptable configuration during normal conditions and postulated accident conditions of transport (e.g. a potentially critical geometry is not prevented), then SNF with stresses exceeding the threshold might be considered to be damaged. This is a case in which damage is not an intrinsic property of the fuel, but depends on the design assumptions. If the applicant can demonstrate that, even if the fuel reconfigures and the cladding does not retain fuel particulate, the regulatory requirements for containment and subcriticality are met, then reconfiguration is not an issue and stress is no longer a measure of damage. Until concerns regarding stress thresholds, effects of reorientation on cladding mechanical properties, and responses of rods with altered mechanical properties are resolved, the use of stress as a determinant of damage cannot be definitively addressed. These concerns and examples illustrate that a nexus exists between the requirements on a fuel rod, the conditions of service, the properties of a fuel rod, and the definition of damage.

6. SUMMARY

Damage is not an intrinsic property of fuel. Classifying a rod as damaged depends on the system, the storage and/or transport conditions, and the requirements on the fuel performance. US Federal Regulations (10 CFR Parts 71 and 72) [3, 4] place minimal direct requirements on the fuel itself. During storage the fuel must not degrade beyond the transport package design requirements. During normal storage or transport the fuel cannot reconfigure. The cask designer can impose indirect requirements and the definition of damaged fuel may change in order to meet the system requirements for containment, confinement, criticality, structural and thermal behaviour. Together, these requirements establish the purpose of the performance of the spent fuel. Fuel that cannot fulfil its defined requirements in the designed storage or transport situation (temperature range and fill gas) should be considered damaged. Damage is defined by the requirements of the system and of the regulations, and fuel may be considered damaged under one scenario but undamaged under another.

The NRC's Spent Fuel Program Office (SFPO) has developed recommended definitions of damaged spent fuel. These are similar to the definitions being proposed by the American National Standards Institute (ANSI). Both allow fuel rods that have pinholes or tight cracks to be considered undamaged. This is based on assumed storage in an inert atmosphere and the need to retain fuel retrievability. But storage in air atmospheres might allow fuel oxidation that promotes gross breaches in the fuel. Damaged fuel assemblies have also been defined in relation to the continued capability of retrieval of fuel from storage and the maintenance of the fuel in a non-critical configuration during transport. Should hydride reorientation significantly reduce the ductility and axial strength of the fuel cladding to the point where the cladding can no longer meet the transport requirements, the cladding stress might need to be considered a characteristic of fuel rod damage. This will depend on whether the requirements on the cladding are mitigated by the use of burnup credit, moderator exclusion or other means.

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DISCUSSION

T. SAEGUSA (Japan): Can you tell us who was responsible for developing the position that you have outlined and what its status is?

R. EINZIGER (United States of America): The position was developed by the material engineers in the Spent Fuel Program Office (SFPO) of the US Nuclear Regulatory Commission. It was generally agreed upon by the staff of the SFPO. An interim staff guidance document is under preparation, but it will have to be approved by all interested parties in the Commission's management before it is issued.

DEVELOPMENT OF APPROACHES FOR THE LONG TERM STORAGE OF DAMAGED NUCLEAR FUEL

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Abstract

The paper provides a summary of the technical and safety issues associated with the management of damaged spent fuel, with special reference to the fuels from the RBMK reactor.

1. INTRODUCTION

In order to safely manage the storage of damaged spent nuclear fuel (DSNF), the following basic safety aspects must be considered:

(a) Nuclear safety;
(b) Radiation safety;
(c) Fire safety;
(d) Ecological safety.

More specifically, the following must be addressed:

(1) Criticality requirements (an effective neutron multiplication factor of 0.95 should not be exceeded for treatment of DSNF in normal operations, abnormal operation conditions and projected accidents);
(2) Reliable removal of residual heat from DSNF in normal operations, abnormal operational conditions and projected accidents;
(3) Minimization of the number of technical operations with DSNF;
(4) Testing, maintenance, radiation control and checking of radioactive contamination of equipment which is being modified and/or new equipment [1];
(5) Safety in decommissioning conditions.

2. CLASSIFICATION OF DEFECTS

2.1. Description of the damaged spent nuclear fuel

A database of operating parameters for spent nuclear fuel should be maintained, containing information on:

(a) Burnup level;
(b) Cooling period;
(c) Isotopic composition;
(d) Date of registration of defect;
(e) Description of defect determination system.

2.2. Classification of mechanical defects

Experience with RBMK fuel elements shows that usually there are the following defects:

(a) Plug break-off in the area of the welding connections near the centre of the core;
(b) Annulus breaks in the fuel container;
(c) Cracks of different configurations and sizes;
(d) Local ballooning with and without crack extension at the crack tip;
(e) Corrosion of fuel cladding — in the area of peripheral electron beam welding connections;
(f) Local or extensive oxidation of the surface of the fuel element with porosity and pitting corrosion.

Defects of types 1-4 are inside the fuel cladding; defects of types 5 and 6 are outside and due to chemical actions [2].

The following are the possible causes of damaged fuel elements:

(1) Development of micro-defects in fuel elements which were not discovered during the production and assembly of the fuel;
(2) Local overheating;
(3) Cracking connected with tension or fatigue and the impact of thermal cycles of the fuel element core, for example, during rapid and substantial changes of power;
(4) Ballooning of fuel cladding and excessive extension of the fuel element due to the accumulation of gas fission products or changing fuel pellet structure;

(5) Chemical reactions of the cladding material with fission products [3].

The main defects of spent nuclear fuel are mechanical, including seal failure of the fuel element cladding. Defects revealed during spent fuel flaw detection include the following:

(i) Fuel assembly drop events causing the removal of rods from the assembly;
(ii) Displacement of fuel elements;
(iii) Division of fuel assembly into two parts;
(iv) Deformation and damage of spacer grids;
(v) Deformation and breakage of fuel elements.

2.3. Control of cladding condition

The maintenance of fuel integrity is facilitated by technical improvements in the reliability of the equipment that controls fuel condition, as well as the proper organization and performance of handling operations, in accordance with operational specifications, and by having a well disciplined staff with appropriate professional qualifications.

The design peculiarities of RBMKs, such as heterogeneous channel disposition of fuel in the reactor core and a boiling forced circulation coolant circuit opened by steam, mean that the fuel cladding failure detection (FCFD) system operates at two levels:

(1) At the first level, continuous checking is carried out for gaseous fission product radioactivity in separated steam at each of four drum separators of the reactor;
(2) At the second level, periodic individual control is carried out by monitoring the condition of each fuel assembly in the reactor core.

Thus the FCFD system performs two main functions:

(i) General checking for the presence and formation of breached fuel elements in the reactor;
(ii) Determination of fuel assemblies with unbreached fuel elements in the reactor channels.

3. ORGANIZATION OF TEMPORARY (INTERIM) STORAGE

3.1. Visual survey for detection of mechanical defects

This involves a system for the remote survey of the condition of the fuel. The objectives of visual control for spent fuel assemblies are:

(a) Assessment of the fuel element surface condition to detect mechanical damage and lack of plugs;
(b) Detection of grid assembly damage and other structural elements of the assemblies;
(c) Determination of the maximum overall size of the spent fuel assembly in the grid assembly plane;
(d) Determination of the clearance between fuel element clusters;
(e) Detection of superfluous items and their location;
(f) Determination of reasons for spent fuel assembly blocking;
(g) Detection of visible assembly flexure;
(h) Detection of missing shank nuts.

Visual inspection of spent fuel assemblies is carried out remotely using TV cameras with colour displays that provide all-round surveillance of the spent fuel assemblies being surveyed. A video control device is used to provide remote control for TV cameras and recording to a video cassette.

3.2. Control of cladding condition

This involves using approved methods for checking the condition of the cladding. Currently the following characterization criterion is used for RBMK reactors. The 'method of three days settling in water of storage tube' is used. If the concentration of any reference nuclides in water withdrawn from a storage tube containing fuel which has been stored in the tube for three days after being withdrawn from the reactor exceeds the values:

I^{131}–1.10^{-3} Ci/kg;
Cs^{137}–1.10^{-4} Ci/kg;
Np^{239}–1.10^{-5} Ci/kg;

then the fuel assembly is considered to be faulty according to the procedure described in Refs [1, 2].

According to Ref. [3] the radionuclide Cs^{137} is the main indicator in the procedure for checking the tightness of fuel.

4. EXTRACTING, HANDLING AND MANAGING DAMAGED SPENT NUCLEAR FUEL

4.1. Problems in extracting and packaging DSNF

Some of the problems that can appear when extracting damaged spent fuel are:

(a) Difficulties in extracting some assemblies;
(b) Damage to fuel during extraction;
(c) Problems with inserting and withdrawing from new storage tubes;
(d) Problems during transport of suspended fuel within the facility.

The methods used for handling the fuel under such circumstances include:

(1) Use of fuel service hot cell;
(2) Underwater systems.

4.2. Technology and procedures required for handling damaged spent fuel at local or centralized facilities

The following precautions should be taken:

(a) During DSNF storage a procedure for the control of the water level in tubes and water replenishment should be developed. This includes a device for the removal of highly active water from tubes without mixing with the water of cooling ponds.
(b) Transport operations involving DSNF should use available standard equipment to the maximum extent possible.
(c) The equipment should be designed to minimize the occurrence of diametrical, longitudinal and bending loads on DSNF during its storage and management.
(d) The possibility of mechanical damage to external surfaces of DSNF during its retrieval and handling should be minimized.
(e) Safe speeds and accelerations for DSNF movements should be specified.
(f) DSNF management activities should be mechanized, automated, and provided with remote means of control to the maximum extent possible.
(g) A technical means for excluding uncontrolled unauthorized movement of equipment for DSNF management should be provided.
(h) The control systems of cranes and other lifting mechanisms for DSNF should exclude the possibility of DSNF being dropped in the case of a

power cut. Electric motors used in the movement of DSNF should have a safe and secure supply of power to avoid failures that could lead to an accident.

(i) Grippers of lifting mechanisms must be reliable and should be designed to ensure the safe handling of DSNF.

(j) The DSNF storage and management system should be designed to withstand locally expected seismic loads.

(k) Equipment should be designed so as to readily facilitate its repair or decommissioning.

4.3. Organization of DSNF drying

Free water is non-evaporated water located either outside SFA or inside the SFA package, or inside separate fuel elements. Absorbed water is water absorbed by vapour in sediments at the surfaces of RBMK SFA. Chemically absorbed water is water chemically bound with sediments at SFA surfaces or on unprotected surfaces of UO_2 fuel pellets in fuel elements. Any free or absorbed water remaining in a spent fuel container can disassociate through radiolysis and oxidation/corrosion and lead to a potential over-pressure condition. Accordingly, water should be removed from the cladding surfaces of damaged fuel. Free water should be removed, e.g. from under cladding surfaces; in addition, the absorbed water should be removed from the surfaces of the spent fuel.

4.4. Measures for spill management

(a) Radioactive material created as a result of spills should be collected and stored safely;

(b) A special place for storage of the fragments should be created;

(c) A careful accounting of the radioactive material contained in the fragments should be performed.

5. CONTAINER REQUIREMENTS FOR LONG TERM STORAGE OF DSNF

The following items must be considered when choosing a container for long term storage of damaged fuel:

(a) Provide an inert storage environment.

(b) Provide for reduced corrosion processes during storage.

(c) Metal corrosion under damp conditions leads to the generation of solid products of corrosion, heat and gas. The mechanism and rate of corrosion of the internal surfaces of the container depend on type of water, temperature and oxidation conditions. The mechanism of container corrosion involves two principal parameters, the presence of condensed or evaporated residual water and the potential availability of gaseous fission products or activation products released to the container due to cracking of the cladding after the fuel is loaded into the container.
(d) Provide two hermetically sealed barriers.
(e) Prevent increase in pressure during storage.
(f) Pressurization in the container could take place either due to gas release from the fuel bundle or as a result of gas generation. Fuel elements contain gaseous fission products, gaseous activation products, helium as a result of initial filling with inert gas during fuel production, and helium as a result of alpha decay. Gas can be released to the container from fuel elements due to diffusion through the metal cladding, or if damage to the fuel cladding occurs during storage. Other gas in the container could be generated by the evaporation of residual water, the corrosion of the container or the contents of the container, and by water radiolysis.
(g) Control mechanical loads on fuel.
(h) Ensure chemical compatibility between storage materials and spent fuel cladding so as to avoid corrosion.
(i) The construction materials should be corrosion compatible and should not contaminate the spent fuel with foreign substances (corrosion products, etc.), which would negatively influence the storage facility functions or degrade the fuel assemblies during the period of storage.
(j) Ensure surface quality is maintained.
(k) Provide operational reliability.
(l) Degradation processes occurring in the container construction materials and in the spent fuel during normal operation and design accidents, including corrosion, creep, fatigue, shrinkage, deterioration, modifications caused by radiation, and other potential processes must be assessed and accounted for.
(m) Optimize costs.
(n) The cost of developing special containers for the long term storage of damaged fuel and the cost of their use, taking into account the special equipment and materials needed for this type of fuel (additional absorbers, water absorbing materials, etc.), should not be excessive. Technical decisions which increase the cost of the container to an unacceptable value for the operator should be avoided if possible.
(o) Ensure stable supply of containers.

6. SAFETY DURING TRANSPORT AND STORAGE OF DAMAGED SPENT FUEL

The following considerations should be addressed:

(a) Events that could affect safety under normal operating and accident conditions should be analyzed and it should be ensured that the system can withstand them.
(b) Ensure that the requirements for criticality safety are complied with. One of the most important problems in damaged fuel storage is ensuring subcriticality in the container when the location of the fuel cladding is uncertain. There may be uncertainty in the description of the fuel matrix condition and the presence residual water and mixtures that can have an effect on subcriticality. In fact, for damaged fuel only the conservative case should be considered in criticality safety analysis, in which the geometrical fuel cladding integrity is assumed to be ruptured and mixed together with a homogenous multiplying medium assumed to be present throughout the storage period.
(c) Provide assurance that the temperature at the surface of the fuel will not exceed limits during storage.
(d) Ensure that fire and explosion risks are minimized. The safety basis for choosing a special container for the storage of damaged fuel must take into account radiolysis–thermolysis of residual water (free and fixed water) that might produce an explosive mixture.
(e) Provide for ongoing monitoring of the fuel and the storage environment. Uncertainty in determining the status of damaged fuel necessitates a reliable monitoring system for the storage condition of this fuel. The system must be able to control a broad set of parameters, including:
    — Pressure inside the container;
    — Composition of the storage environment of fuel;
    — Temperature inside and outside of the container.

    Intervention criteria must be developed, i.e. the boundary values for safety which, if exceeded, would invoke corrective measures (for example depressurization using special valves and devices) or a complete re-storing of the damaged fuel.
(f) Provide assurance of compliance with radiation safety requirements in normal operation for emergency situations, design and out of design accidents.

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TRENDS IN DRY SPENT FUEL STORAGE AND TRANSPORT IN GERMANY AND THE UNITED STATES OF AMERICA

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Abstract

In Germany and the USA dry storage is being used increasingly to bridge the time period until final disposal concepts can be realized. In parallel, the burnup of spent fuel is increasing and traditional burnup limits are being reached and exceeded. The paper describes the present situation with respect to fuel developments and the implications for long term storage and transport, describes current research, prompted by regulatory concerns, to evaluate the long term behaviour of high burnup spent fuel, and presents the first results of long term creep tests.

1. INTRODUCTION

The AREVA company provides nuclear fuel solutions worldwide for the entire fuel cycle, from uranium mining to radioactive waste disposal. As part of that effort, the company supports a programme to track global trends in spent nuclear fuel storage and transport. This paper reports on country specific licensing trends and related data and research in Germany and the USA relevant to the implementation of dry storage technologies.

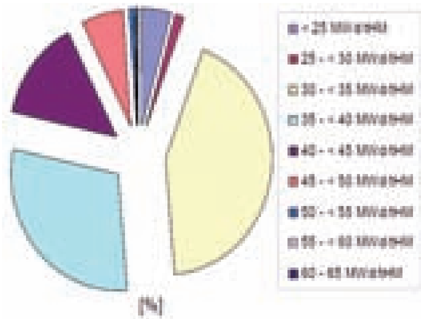


FIG. 1. Burnup distribution of the fuel assemblies currently stored in CASTOR™ V/19 and V/52 casks in Germany.

2. BACK END CYCLE

2.1. Germany

Germany's revised energy act of 2002 prohibits the shipment of spent nuclear fuel to reprocessing plants and requires that it be disposed of in a final repository. To comply with this law and to ensure further nuclear plant operation the reactor operators have decided to construct on-site facilities for dry cask storage to store spent fuel assemblies until a final repository is available. There are three technical concepts for on-site storage: storage buildings, a storage tunnel, and interim storage areas. While storage buildings and the tunnel are envisaged for a licence duration of 40 years, the interim storage areas are mainly foreseen for a licensing duration of 5–8 years to bridge the time until the storage buildings are commissioned. Thirteen facilities are planned or under construction with a total of about 1400 places for casks. At the present time, four storage buildings are in operation and four interim storage areas are in use. A total of 100 CASTOR® V/19 and V/52 casks, equivalent to 2362 fuel assemblies, have been successfully loaded. The burnup of 90% of the fuel is below 45 MW·d/kg HM, but some fuel assemblies have burnups of up to 65 MW·d/kg HM (Fig. 1).

During the last 20 years the fuel management strategy has been characterized by a continuing effort to reach higher burnups by increasing the U-235 enrichment and, in the case of the MOX fuel assemblies, the fissile Pu content. Figure 2 shows the maximum fuel assembly burnup of the leading refueling batch versus the year of discharge from the reactor [1]. At the present time, the maximum discharge burnup of commercial fuel manufactured by AREVA NP GmbH ranges between 50 and 55 MW·d/kg HM. To gain irradiation experience

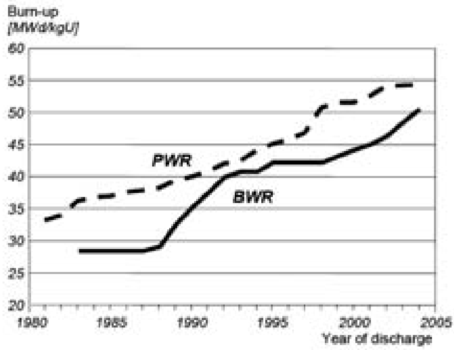


FIG. 2. Maximum burnup of the leading refueling batch as a function of the year of discharge.

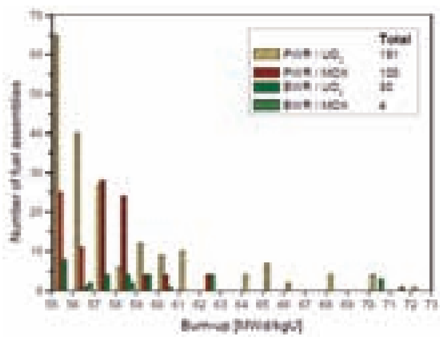


FIG. 3. Burnup statistics of fuel assemblies from AREVA NP GmbH with burnups beyond 55 MW·d/kg HM in 2006.

at even higher burnups a limited number of fuel assemblies have been irradiated up to burnups of about 70 MW·d/kg HM. The statistics for fuel assemblies that were manufactured by AREVA NP GmbH and reached burnups beyond 55 MW·d/kg HM in 2006 are shown in Fig. 3. Most of the high burnup fuel is of the PWR type, and about a third of this is MOX fuel. UO_2 BWR fuel assemblies have a share of about 10%. It has to be noted that fuel assembly burnups up to 72 MW·d/kg HM can be reached for both PWRs and BWRs with modern highly corrosion resistant cladding materials. Figures 4 and 5 show the maximum oxide thicknesses on PWR and BWR claddings as a function of the fuel rod burnup. It can be seen that for a maximum batch discharge burnup of 55 MW·d/kg HM, equivalent to a rod burnup of about 60 MW·d/kg HM, the corrosion layer thickness of DX ELS0.8b remains below

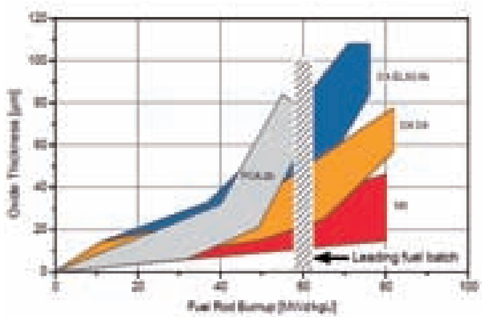


FIG. 4. Corrosion behaviour of modern PWR cladding materials up to very high burnups.

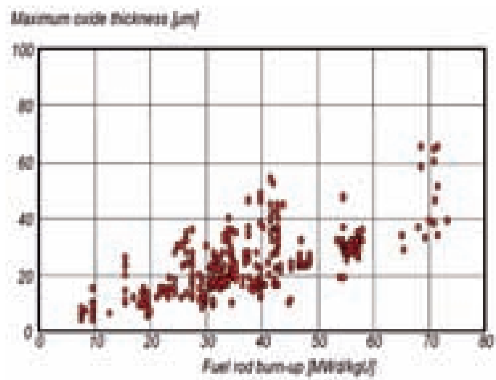


FIG. 5. Corrosion behaviour of modern BWR cladding materials.

about 80 μm, for DX D4 below 50 μm, and for M5® below 40 μm. For BWR cladding a similar burnup leads to maximum oxide layers of about 60 μm.

2.2. USA

The USA uses a 'once through' nuclear fuel cycle and spent fuel is currently being stored at the reactor site. A licence application for the national spent fuel repository at Yucca Mountain is still in preparation. There is active research on reprocessing, advanced reactors, and advanced fuel cycles to reduce the amount and radiotoxicity of radioactive waste. A centralized private spent fuel storage facility has been licensed but is not yet under construction.

Although the situation in the USA may appear to be static, some important changes are occurring. Firstly, because there is no licensed disposal site and because of the limitations of fuel pool capacity, the use of at-reactor dry fuel storage is increasing. Sixty-seven of the 104 licensed reactors currently have contracts for dry storage facilities, and over 9000 Mt U of fuel is in storage. By 2010, the number of reactors with dry storage is expected to increase to 85, and the amount of fuel to 20 000 Mt U. Secondly, because of the increase in the inventory of spent fuel it is inevitable that there will eventually be a large increase in the amount of spent fuel transport. The destination of this transport could be a geological repository, a long term surface storage site or a reprocessing facility. Thirdly, the spent fuel being discharged today is different from that discharged in the past because new cladding alloys are being used and fuel burnups are continuing to increase. All three of these trends have prompted renewed regulatory scrutiny.

3. LICENSING

In relation to dry cask storage systems the main safety concerns are to ensure subcriticality during storage and retrievability after storage. Safety is achieved by maintaining the integrity of the fuel assembly structure and preventing systematic cladding failures. These are basic and common requirements from which country specific rules have been derived.

In Germany, criteria for the dry storage of spent fuel assemblies were promulgated by the Reactor Safety Commission in 2002 [2]. It is stated in these criteria that the mechanical integrity of the fuel assembly structure must be maintained during handling, storage, transport for final storage, and discharge operations. Systematic cladding failures have to be avoided by limiting the stress and strain on the material under consideration. Defective fuel rods need special treatment and/or confinement. The present engineering approach to ensure cladding integrity is to impose limits of 1% circumferential plastic strain and 120 MPa tangential (hoop) stress. These values limit thermal creep degradation and hydride reorientation under dry storage conditions.

For the handling of individual damaged fuel rods, special rod capsules and canisters have been developed (Fig. 6). The capsules can be drained and stored in canisters with the dimensions and handling properties of fuel assemblies. The canister itself will be stored in the spent fuel pool and in dry casks. Licensing of the capsules presently covers only handling and storage in the reactor pool. Licence applications for storage in dry casks and shipment are in preparation.

At the present time, CASTOR® V/19 and V/52 casks are licensed in Germany. The CASTOR® casks are licensed for a maximum fuel assembly

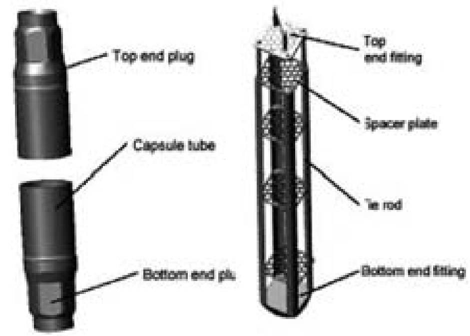


FIG. 6. Rod capsule and rod canister to store damaged fuel rods.

burnup of 65 MW·d/kg U (UO_2) and 55 MW·d/kg HM (MOX). These licences cover most of the burnups currently being reached for UO_2 and MOX fuel. Compliance with the design limits for cladding stress and strain can usually be demonstrated even for fuel with higher burnups. However, this depends on the loading inventory of the cask since the design criterion on creep strain is very sensitive to the maximum cladding temperature.

A licence for the TN®24E cask for 21 PWR fuel assemblies has been applied for and approval is expected to be granted by the end of 2006. The TN® 24 E cask allows the loading of a batch of UO_2 spent fuel with 60 MW·d/kg U average burnup and 5 years of cooling. Specific loading plan assessment procedures will allow the loading of single spent fuel rods with much higher burnup (up to 75 MW·d/kg U) or much shorter cooling times (minimum of 2 years). The minimum cooling time required prior to allowing a MOX fuel to be loaded into a dual purpose cask is much longer than for UO_2. The TN® 24 E cask design will hold configurations of 13 UO_2 + 8 MOX and 4 UO_2 + 17 MOX. Sixty-four casks of the TN®24 family have already been loaded in Belgium and Switzerland.

In the USA, regulatory guidance recommends that the cladding temperature should be kept below 400°C and that thermal cycling should be minimized during fuel loading and drying operations [3]. The temperature limit serves to prevent creep failures and the restriction on thermal cycling serves to prevent hydride reorientation. Cask manufacturers typically ensure compliance with the temperature recommendation by specifying maximum thermal loads. The guidance is based on time in storage, cooling time after discharge, fuel rod internal pressure, burnup, and material properties of Zircaloy cladding.

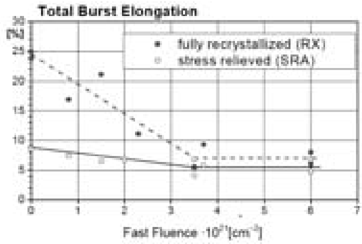


FIG. 7. Total burst elongation of irradiated RX and SRA Zircaloy materials under burst test conditions.

The areas of greatest regulatory concern appear to be (1) hydride reorientation, (2) initial fuel condition after irradiation, (3) mechanical properties of irradiated cladding, and (4) fuel drying. ‘Initial fuel condition’ refers to the condition of fuel rods when put into dry storage or transported. It is affected not only by the design of the fuel but also by irradiation conditions, such as coolant chemistry and power history. It is therefore necessary to take into account the fuel condition on the basis of reactor operating records and examinations of discharged fuel rather than predicting it on the basis of the design and ‘as-manufactured’ state of the fuel. The other three items are discussed below.

4. RESEARCH ACTIVITIES

4.1. Review of the database

The trend to higher discharge burnups has required extension of the database for cladding integrity assessment. In the 1980s, rod internal pressures and cladding fluence levels were lower than those of more recent fuels and cladding behaviour could be assessed by burst tests. Burst tests feature short testing times due to high stresses in the yield stress regime of the material. Burst test results for stress relieved (SRA) and fully recrystallized (RX) cladding materials are shown in Fig. 7. Under these test conditions at low fluences, RX cladding reveals a higher elongation than SRA cladding. At increased fluences, the distinction between SRA and RX almost disappears [4].

Higher burnups and higher hydrogen levels have led to the need for more realistic creep rupture tests at lower stress levels and at storage temperatures of 300–400°C [5]. Creep rupture tests are devised to analyze the failure behaviour of samples that have been subjected to stresses below the yield stress for long

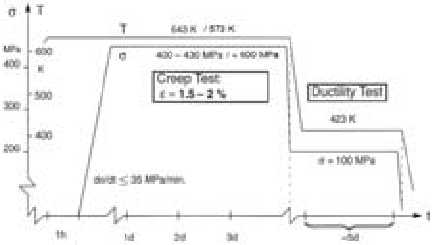


FIG. 8. Testing sequence of the creep rupture tests.

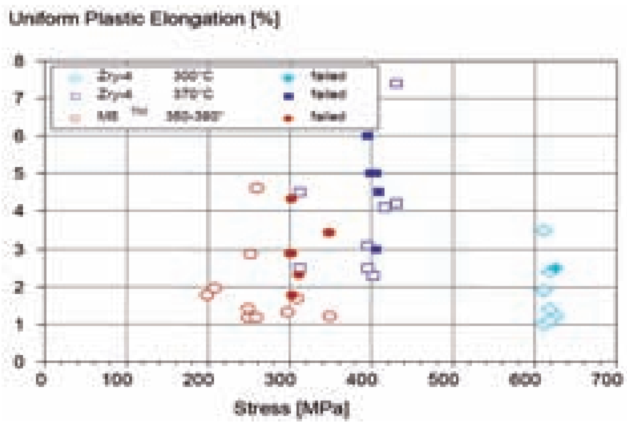


FIG. 9. Failure behaviour of Zircaloy-4 and M5® cladding under creep rupture conditions.

periods of time. Deformation of the cladding occurs by thermal creep rather than by the nearly instantaneous plastic deformation characteristic of a burst test. The creep rupture behaviour was experimentally determined on irradiated, corrosion optimized Zircaloy-4 cladding from fuel with burnups of 54–64 MW·d/kg U. The tests were performed in two steps: the first step was a creep test with a projected elongation of 1.5–2%, while the second step was meant to induce maximum hydrogen embrittlement. The fuel was cooled under a circumferential stress of about 100 MPa and subsequently held at low temperatures (150°C) for several days (Fig. 8). The tests yielded plastic strains of at least 1% without failure at stresses up to 620 MPa, as shown in Fig. 9. All tests showed that even under unfavourable conditions with radial hydrogen precipitation, there was no adverse effect on the Zircaloy cladding behaviour.

FIG. 10. (a) Design of the gas filled rod samples for long term testing.

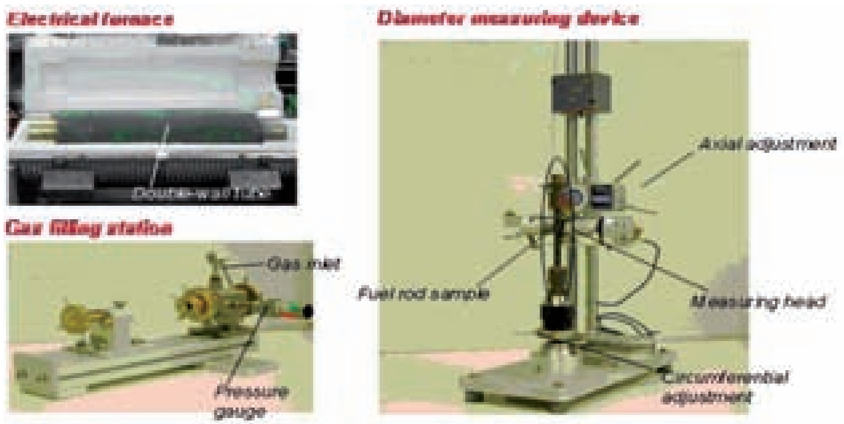


FIG. 10. (b) Equipment for long term creep testing and diameter measuring.

4.2. Long term creep testing

The most realistic way to assess the long term behaviour is to perform creep tests at stresses and temperatures that reflect the cask storage conditions. To this end, short gas filled rods have been used to simulate the cladding behaviour in the central region of the storage cask. AREVA NP is performing such tests on its commercial PWR and BWR cladding materials, such as Extra low tin (ELS) Duplex/Zircaloy-4, M5™ and Zircaloy-2/LTP.

Figure 10 shows the design of the specimen, the furnace and the equipment to fill the specimen with gas and to measure its diameter after removal from the furnace. It has to be noted that gas filled specimens establish a constant circumferential cladding stress due to wall thinning in combination with diameter increase. Initial results for 673°K and 130 MPa circumferential stress are shown in Fig. 11. It is seen that stress relieved (Zry-4 SRA) and recrystallized (Zry-2 RX) cladding behave quite differently under thermal

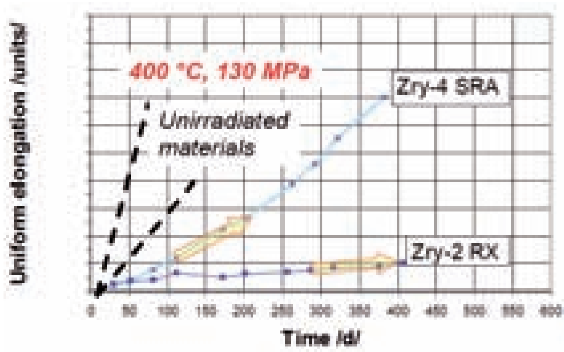


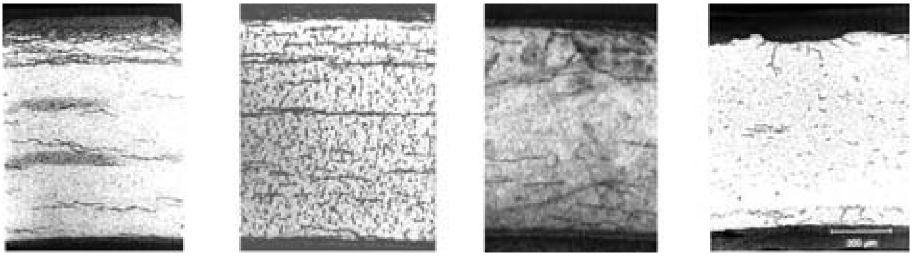
FIG. 11. Creep behaviour of irradiated and unirradiated Zircaloy-4 SRA and Zircaloy-2 RX materials under long term test conditions.

creep conditions. However, contrary to the burst test results, the SRA cladding exhibits a much faster secondary creep than the RX material. Therefore, the Zircaloy-4 SRA material reaches 1% strain earlier than the Zircaloy-2 RX material.

The basic idea of the long term experiments is to measure the time required to burst the specimen under a given internal gas pressure. In the case of steady state creep, the rupture time, t_R, is given by $t_R = \varepsilon_F/\dot{\varepsilon}$ where ε_F is the diametral strain at failure and $\dot{\varepsilon}$ is the creep rate, which is assumed to be constant for $0 < t < t_R$ [6]. The creep rupture experiments [5] yielded a steady state creep rate of 1E-7 s^{-1} (3% strain and t_R = 60 h) for Zircaloy-4 at 643°K. From Fig. 11, a steady state creep rate of about 2E-9 s^{-1} for Zircaloy-4 SRA and 2E–10 s^{-1} for Zircaloy-2 RX can be derived. By comparing the creep rates, a 50 times longer lifetime, equivalent to 125 days, can be estimated for the Zircaloy-4 SRA material assuming a comparable rupture strain.

4.3. Hydride reorientation

The commercial cladding materials that are candidates for dry storage cover a wide range of hydrogen concentrations. Whereas the Duplex ELS-type may show hydrogen concentrations of several hundreds of ppm, Zircaloy-2 may reach 100–200 ppm, and M5™ will remain below 100 ppm [7] up to the highest burnups foreseen. The hydrogen distributions in different types of PWR and BWR claddings are shown in Fig. 12. It is important to note that, in the case of high hydrogen concentrations, most of the hydrogen is concentrated in an outer hydride rim under irradiation conditions. Such a distribution is seen in Fig. 12(a), which shows Duplex cladding with several hundreds of ppm. During cooling under stress and uniform temperatures the solubility of



(a) (b) (c) (d)

FIG. 12. (a) PWR cladding with outer hydride rim, (b) PWR cladding after cooling down under 100 MPa stress, (c) PWR M5® cladding with a burnup of 70 MW·d/kg HM, (d) BWR Zircaloy-2/LTP cladding with inner liner.

hydrogen limits its radial redistribution. This means that most of the hydride rim remains undissolved and thus the absolute amount of hydrogen that is available on the inner side of the cladding, where the stress is highest, is limited. The samples of Fig. 11 had local burnups of 63–65 MW·d/kg U and hydrogen contents of around 100 ppm for Zircaloy-2 and several 100 ppm for Zircaloy-4. Given that the rim retains most of the hydrogen, high burnup duplex samples with even thicker oxide layers are very similar with regard to the hydrogen content in the inner part of the cladding. Thus the tested duplex samples represent rods with even higher burnups. For M5™ cladding, the hydrogen situation is completely different. Figure 12(c) shows a typical cross-section of irradiated M5™ cladding. Although the fuel rod was irradiated to an average burnup of 70 MW·d/kg U, the hydrides are still sparse, with a hydrogen concentration of only about 70 ppm. As shown in the figure, the hydrides are primarily circumferential. M5™ shows corrosion rates and hydrogen pickup fractions that are consistently low in reactors worldwide.

The effect of hydrogen on mechanical behaviour has been studied for many years. From burst tests it is known that hydrogen can reduce the tensile strength and ductility as compared with hydrogen free material, especially when the hydride orientation is perpendicular to the applied stress. However, at temperatures above about 473°K, the effect of hydrogen on the ductility is less pronounced than it is at room temperature, and in irradiated materials the irradiation effect is the dominant influence on ductility up to high hydrogen concentrations. In the long term creep tests (Fig. 11) it is necessary to cool the gas filled sample to measure its diameter. The cooling process, which takes about one day, is considered to constitute a severe test but, until now, no striking evidence of adverse effects due to radial hydrogen precipitation has been found in these tests.

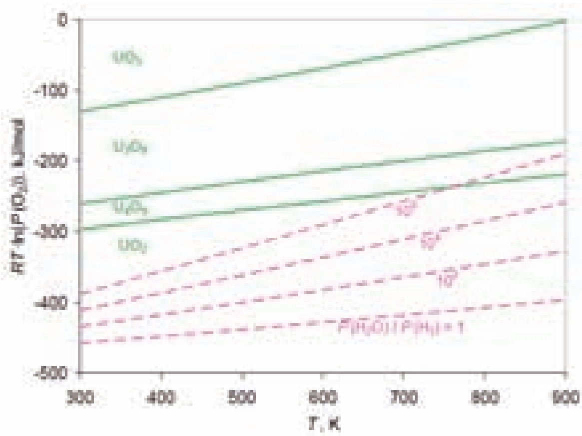


FIG. 13. Oxidation stability diagram for oxidation of uranium and for H_2O–H_2 equilibrium.

4.4. Oxidation of UO_2

Casks for transport and dry storage of spent nuclear fuel are typically loaded in the spent fuel pool. Although procedures have been defined for fuel drying there is the potential that residual water will nevertheless be present in the cask. Occluded water, such as that in the dashpots of the guide tubes, in the annuli between burnable poison rods and guide tubes, and in failed, waterlogged fuel rods, is difficult to remove. Water is of regulatory concern in part because it can serve as a moderator and in part because water can, in principle, serve as a source of oxygen and cause fuel oxidation. Oxidation of UO_2 to U_3O_8 causes a significant volume expansion, which could result in the splitting of fuel cladding.

Splitting of cladding has been observed in defective fuel rodlets that were tested in an air steam environment at 175°C. It is not known whether a similar splitting will occur in a helium steam environment, which would be expected in a sealed storage or transport cask. Simple thermodynamic considerations suggest that it will not. Figure 13 shows an oxidation stability (Ellingham) diagram for the oxidation of uranium. For a given absolute temperature T, equilibrium between two oxides of uranium specifies a particular value of the oxygen potential $RT{\cdot}\ln(P(O_2))$, where R is the gas constant and $P(O_2)$ is the partial pressure of oxygen in bars. These equilibria are shown as solid lines. Also shown, in dashes, are lines of constant $P(H_2O)/P(H_2)$. Oxidation of fuel would require reduction of some steam to hydrogen. However, the figure shows that for temperatures up to about 750°K, UO_2 is stable even for large values, 10^6 or greater, of $P(H_2O)/P(H_2)$. The implication is that even a small amount of fuel oxidation will result in a partial pressure of hydrogen, $P(H_2)$, sufficient to prevent further oxidation. Although this argument is encouraging, experiments would ultimately be necessary to determine whether non-equilibrium effects such as radiolysis of water could allow fuel oxidation to occur.

5. CONCLUSIONS AND PERSPECTIVES

Dry storage already plays an important role in spent fuel management, and that role will increase over the next several years. Storage and transport casks have an excellent safety record, with no recorded releases of radioactive material. This performance record is expected to continue, even as the number of casks increases into the thousands.

The initial long term creep test results are consistent with typical creep rupture behaviour, i.e. long lifetimes and elevated strain levels can be obtained with low creep rates. On the other hand, the role of hydride reorientation on sample failure appears to be less pronounced than expected. With regard to integrity assessment there is no evidence that further restrictions are needed, even at the highest burnups attainable under commercial operation.

The creep tests also confirmed that, in spite of very long annealing times, high burnup Zircaloy cladding always behaves differently to unirradiated material with regard to creep and rupture behaviour. Whereas the creep rate is defined by stress and temperature for a specific material, creep rupture depends on an additional parameter such as strain rate or holding time, and therefore requires considerably greater experimental effort for its evaluation. Future research should therefore focus on the creep rupture parameters that control the maximum strain attainable under low stress conditions.

Fuel oxidation remains a challenge. Current cask drying procedures appear to be fairly stringent, but there is little technical basis for determining whether they might be relaxed or should be further tightened. Experiments on irradiated fuel in a helium steam environment may be necessary.

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HIGH BURNUP FUEL CLADDING TUBE PROPERTY TESTING FOR THE EVALUATION OF SPENT FUEL INTEGRITY DURING INTERIM DRY STORAGE

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Abstract

Tests to evaluate fuel integrity in interim dry storage carried out by the Japan Nuclear Energy Safety Organization (JNES) comprised hydride effect evaluation testing, irradiation hardening recovery testing and creep testing. In each test, both BWR and PWR irradiated cladding tubes were used as test material. Up to 2005, hydride effect evaluation testing has been performed using both 5 cycle irradiated Zry-2 cladding for BWR 50 GW·d/t type fuel and 3 cycle irradiated Zry-4 cladding for 39 and PWR 48 GW·d/t type fuel. The behaviour of hydride reorientation in relation to radial direction and the mechanical properties of material after hydride reorientation treatment were investigated for each irradiated cladding tube material.

1. JNES TEST PLAN

The Japan Nuclear Energy Safety Organization (JNES) has been planning and performing tests in order to evaluate fuel integrity in interim dry storage. Table 1 shows the test schedule. By 2003, thermal creep testing and irradiation hardening recovery testing had been performed using irradiated cladding tubes and the creep equation for BWR (50 GW·d/t type) and PWR (48 GW·d/t type) fuel cladding had been established [1, 2]. Since 2004, JNES has been carrying out the tests using irradiated cladding tubes, in which the effects of hydride reorientation have been investigated for BWR and PWR fuel cladding of up to 55 GW·d/t. This paper reports on the results of hydride effects evaluation testing obtained during 2004–2005.

TABLE 1. JNES TEST SCHEDULE

		2000	2001	2002	2003	2004	2005	2006
Survey and Planning								
Creep Test	>Creep test	PWR 48G WP/t, BWR 50G WP/t					PWR 55G WP/t, BWR 55G WP/t	
	>Creep rupture test	PWR 48G WP/t, BWR 50G WP/t						
Hydride Effects Evaluation Test	>Hydride reorientation test >Mechanical property test						PWR 48G WP/t, 55G WP/t BWR 50G WP/t, 55G WP/t	
Irradiation Hardening Recovery Test		PWR 48G WP/t, BWR 50G WP/t 330deg.C–420deg.C					<330deg.C	

2. EXPERIMENTAL PROCEDURE IN HYDRIDE EFFECT EVALUATION TEST

2.1. Test material

The test materials are summarized in Table 2. Up to 2005, hydride effect evaluation testing has been performed using both 5 cycle irradiated Zry-2 cladding for BWR 50 GW·d/t type fuel and 3 cycle irradiated Zry-4 cladding for PWR 48 GW·d/t type fuel. In this paper, the results of Zry-4 cladding for PWR 39 GW·d/t type fuel are also reported.

2.2. Hydride reorientation test

The hydride effect evaluation test is composed of a hydride reorientation test and a mechanical property test. The hydride reorientation treatment (HRT) method is shown in Fig. 1. In the hydride reorientation testing, the HRT was performed as follows: the hoop stress in the cladding tube specimen was applied by inner pressure; the specimen temperature was held for 30–60 minutes at the HRT temperature in the furnace to dissolve the hydride in the cladding; then the specimen temperature was decreased to around room temperature to precipitate the hydride. In most test conditions, the hoop stress in the cladding was held constant as explained in Fig. 1(b), while in one test, hoop stress was decreased with temperature. The degree of hydride reorientation to radial direction was evaluated from metallography of the specimen after HRT.

The hydride reorientation test matrix is shown in Table 3. In the test for BWR fuel cladding, hoop stress conditions were varied from 0 to 100 MPa and

TABLE 2. TEST MATERIALS

Fuel cladding type	Material	Heat treatment	Burnup (GWd/t, rod average)	Hydrogen content (ppm)
BWR (50 GWd/t type)	Zry-2 (with Zr liner)	RX[a]	~48	~133 – ~264
PWR(48 GWd/t type)	Zry-4	SR[b]	~44 – ~46	~50 – ~210
PWR(39 GWd/t type)	Zry-4	SR[b]	~39	~60 – ~100

[a] RX: Re-crystallized annealing.
[b] SR: Stress Relieved annealing

(a) Test apparatus for HRT (b) Temperature and hoop stress in HRT

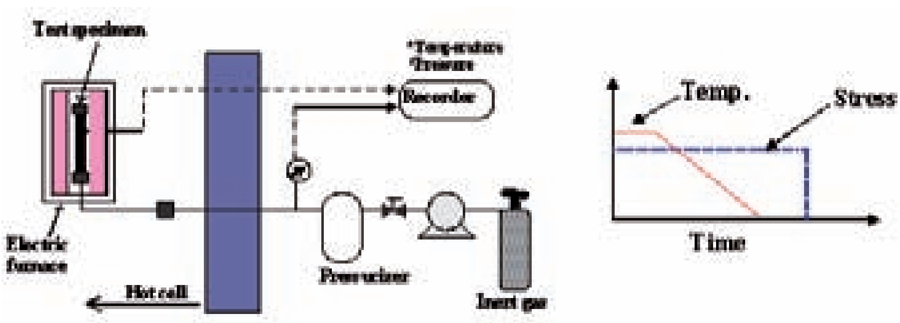

FIG. 1. Hydride reorientation treatment (HRT) method.

the temperatures from 250°C to 400°C. In the test for PWR fuel cladding, hoop stress conditions were varied from 0 to 130 MPa and temperatures from 250 to 340°C. The effect of cooling rate on hydride precipitation was also tested.

2.3. Mechanical property test

The mechanical property tests, which are composed of a ring compression test and a longitudinal tensile test, were performed after hydride reorientation testing in order to evaluate the circumferential and longitudinal mechanical properties of hydride reorientation treated specimens. Figure 2 shows a rough diagram of the test methods. In the ring compression test, ring specimens were prepared from the cladding tube after hydride reorientation testing. The ring specimen was compressed in the radial direction with a crosshead speed of

TABLE 3. HYDRIDE EFFECT EVALUATION TEST MATRIX (HRT CONDITIONS)

(a) Ring compression test

Temp. (°C)	Cooling rate (°C/h)	Hoop stress (MPa) 0	16	28	40	70	100
400	30	B	B	B	B	B	—
300	30	D	B	B	B	D	B
250	30	B	—	B	A	—	—

(b-1) Zry-4 (PWR 48 GW·d/t type)

Temp. (°C)	Cooling rate (°C/h)	Hoop stress (MPa) 0	85	100	115	130
340	30	—	—	A	—	B
300	30	B	A	B	B	B
275	30	A	—	A	B	B
250	30	—	—	A	—	B
300	3	B	—	—	B	E
300	0.6	—	—	—	B C	—

(b-1) Zry-4 (PWR 48 GW·d/t type)

Temp. (°C)	Cooling rate (°C/h)	Hoop stress (MPa) 0	85	100	115	130
300	30	A	—	A	—	A
275	30	A	—	A	—	A

A: Hydride reorientation test (stress const.).
B: Hydride reorientation test (stress const.) + ring compression test after HRT.
C: Hydride reorientation test (stress decreasing with temp.) + ring compression test after HRT.
D: Hydride reorientation test (stress const.) + ring compression test + longitudinal tensile test after HRT.
E: Hydride reorientation treatment (stress const.) + longitudinal tensile test after HRT.

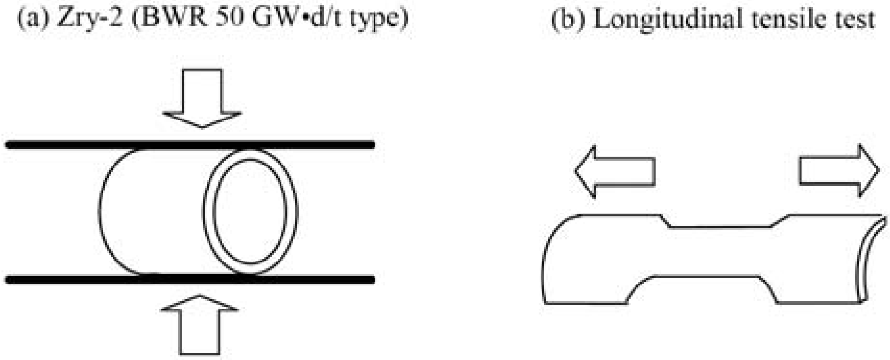

FIG. 2. Mechanical property test method.

about 2 mm/min at room temperature. In the longitudinal tensile test, tensile specimens were prepared from the cladding tube after HRT. The strain speed was about 8.3×10^{-5}/s. The HRT conditions of mechanical property tested specimens are also shown in Table 3. Both the ring compression test and the longitudinal tensile test were performed at room temperature.

3. RESULTS AND DISCUSSION IN HYDRIDE EFFECT EVALUATION TESTS

3.1. Hydride reorientation test

3.1.1. Zry-2 cladding (BWR 50 GW·d/t type)

Figure 3 shows the metallography of BWR 50 GW·d/t type Zry-2 cladding after hydride reorientation treatment (HRT). As shown in Figs 3(a) and (c), a significant amount of hydride reorientation to the radial direction was observed at an HRT hoop stress of 70 MPa at 400°C. However, Figs 3(a) and (b) indicate that a small degree of hydride reorientation to the radial direction occurred at a HRT hoop stress of 70 MPa at 300°C.

In order to clarify the HRT temperature and hoop stress dependence of hydride reorientation on radial direction, the degree of radial hydride orientation was evaluated using Fn(40°, length), which is defined in Eq. (1). Here, hydrides with lengths of over 16 micrometres were analyzed by metallography (not including the Zr liner nor the interface between the Zr liner and the Zry-2 matrix).

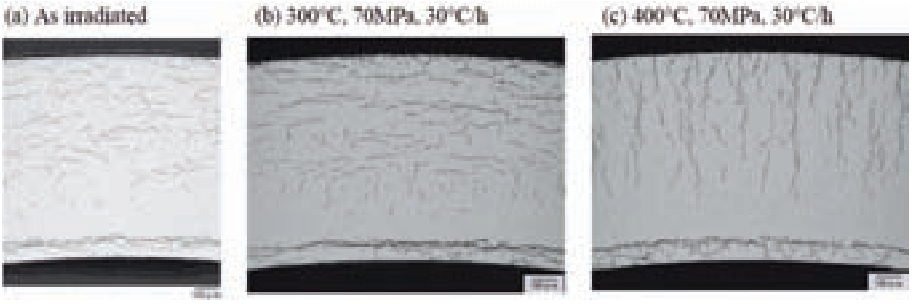

FIG. 3. Metallography of irradiated BWR 50 GW·d/type Zry-2 cladding after HRT

$$\text{Fn}\,(40°, \text{length}) = \frac{\text{Sum of the length of hydrides in radial direction} \pm 40°}{\text{Sum of the length of total hydrides}} \tag{1}$$

Figure 4(a) shows the relationship between Fn(40°, length) and the hoop stress at each HRT temperature. No increase of Fn(40°, length) was observed at 250°C. At 300°C, a slight increase of Fn(40°, length) was observed at HRT hoop stresses of both 40 MPa and 70 MPa, and a considerable hydride reorientation occurred at a HRT hoop stress of 100 MPa. At 400°C the Fn(40°, length) increased with the hoop stress within the tested conditions.

To understand the radial hydride reorientation phenomena more clearly, Oohama et al. proposed [3] the effective hydride radial reorientation factor, or 'Fe' as an index which describes the ratio of radially reorientated hydride in dissolved hydrogen at HRT temperature, while the Fn(40°, length) is defined to describe the ratio of radial hydride in total hydrogen after HRT. In this paper, the same type of index as defined by Oohama [3] is defined in Eq. (2) as Fe(40°).

$$\text{Fe}(40°) = \frac{\text{Content of precipitated hydrides in radial direction} \pm 40°}{\text{Content of dissolved hydrogen at HRT temp.} (= C_{HD})} \tag{2}$$

where,

Content of precipitated hydrides in radial direction ±40° = $C_H \times$ Fn(40°, length) – $(C_H - C_{HD}) \times F0$

C_H = Total hydrogen content of the specimen

C_{HD} = Dissolved hydrogen content at HRT temperature

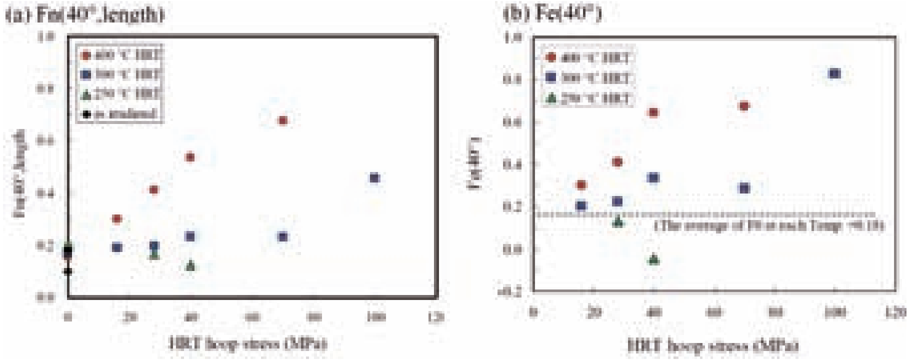

FIG. 4. Correlation between hydride orientation and hoop stress in HRT for irradiated BWR Zry-2 cladding (cooling rate: 30°C/h).

F0 = Fe(40°) at each HRT temperature in HRT hoop stress condition of 0 MPa (heat treatment only)

If C_H is more than C_{HD} in Eq. (2), C_{HD} is equal to the terminal solid solubility of cladding material.

If C_H is less than C_{HD} in Eq. (2), C_{HD} is equal to C_H and Fe(40°) is equal to Fn(40°, length).

The correlation between Fe(40°) and the HRT hoop stress at each HRT temperature is shown in Fig. 4(b). At 400°C, the radial hydride reorientation behaviour in Fe(40°) was similar to that in Fn(40°, length), because the hydrogen in most cladding specimens is considered to dissolve completely at HRT temperature. At 300°C, the radial hydride reorientation tendency is more clear in Fe(40°) compared to that in Fn(40°, length), namely that some increase of Fe(40°) is observed at HRT hoop stresses of 40 MPa and 70 MPa, and a considerable hydride reorientation occurs at an HRT hoop stress of 100 MPa.

3.1.2. Zry-4 cladding (PWR 39, 48 GW·d/t type)

Figure 5 shows the metallography of PWR Zry-4 cladding after HRT. Figures 5(a), (b) and (d) show that some increase of radial hydride occurred at HRT hoop stresses of 115 MPa and 130 MPa at 300°C.

The effect of cooling rate on hydride precipitation can be compared between Figs 5(b) and (c). These results were obtained at the same HRT temperature and hoop stress condition (300°C, 115 MPa). The length of

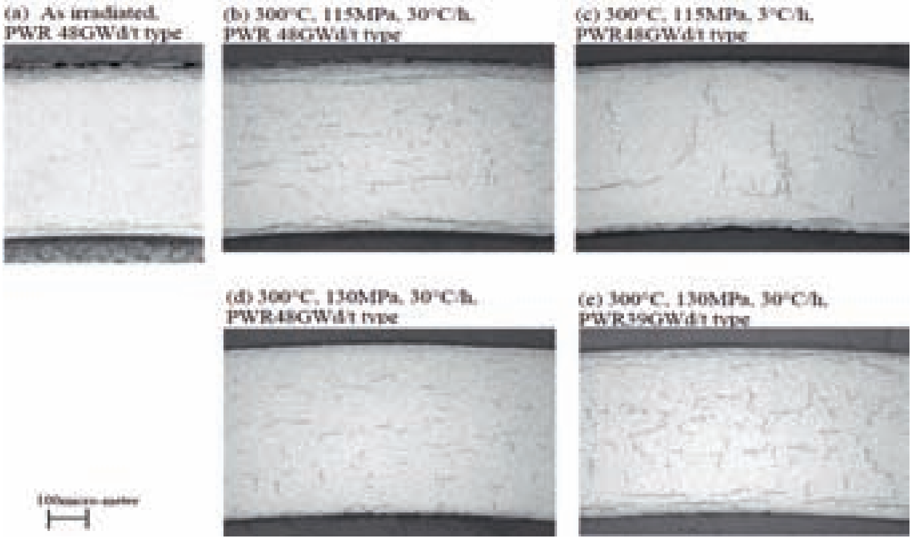

FIG. 5. Metallography of PWR Zry-4 cladding after HRT.

hydride at the 3°C/h cooling rate seems to be longer than that at the 30°C/h cooling rate.

In Figs 5(d) and (e), the effect of material property difference between 39 GW·d/t and 48 GW·d/t type cladding on radial hydride reorientation is shown. Under the same HRT conditions (300°C, 130 MPa), the 39 GW·d/t type cladding shows larger radial hydride reorientation.

In order to clarify the radial hydride reorientation phenomena Fe(45°), the same type of index as used in Zry-2 test Fe(40°) by Oohama [3] was defined. In the test for Zry-4 cladding, Fe(45°) was used to compare the test results with literature data [4] in which Fe(45°) is used as the index. The definitions of Fe(45°) and Fn(45°, length) are the same as Fe(40°) and Fn(45°, length) in Eqs (1) and (2), except that the threshold angle is 45° in Fe(45°) and Fn(45°, length). Fe(45°) was calculated from the hydrides except those precipitated near the outer and inner surface area, because locally concentrated hydride precipitation was observed near the outer and inner surface in Zry-4 cladding. Here, assuming that the hydrogen in the mid-wall area dissolved completely because of its small hydrogen content, Fe(45°) was considered to be equivalent to Fn(45°, length) in the mid-wall area.

The correlation between Fe(45°) and HRT hoop stress, calculated based on the measured wall thickness at each HRT temperature for 48 GW·d/t and 39 GW·d/t type cladding is summarized in Figs 6(a) and (b). In Fig. 6(a), Fe(45°) increases at a hoop stress of over 100 MPa at 300°C. At hoop stresses of

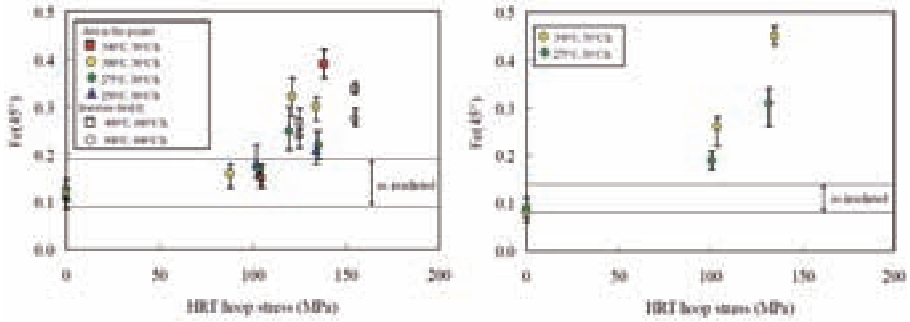

FIG. 6. Correlation between hydride orientation and hoop stress in HRT for irradiated PWR Zry-4 cladding.

100 MPa, the radial hydride reorientation increases with temperature in the range 250°C to 300°C. Comparing Figs 6(a) and (b), radial hydride reorientation tends to occur much more for 39 GW·d/t type cladding than for 48 GW·d/t type cladding.

3.1.3. Discussion

BWR 50 GW·d/t type Zry-2, PWR 48 GW·d/t type Zry-4, and PWR 39 GW·d/t type Zry-4 cladding tubes show different behaviour in hydride reorientation with radial direction. The annealing process during cladding production is different for Zry-2 and Zry-4, and the radial texture of the material is also different for 50 GW·d/t type Zry-2, 48 GW·d/t type Zry-4, and 39 GW·d/t type Zry-4 cladding tubes. These differences may affect the hydride reorientation properties of the cladding materials.

As regards the effect of cooling rate on radial hydride reorientation, Fig. 5 shows that the length of hydride tends to be longer for relatively slow cooling rates, while the hydride orientation for slow cooling rates showed a larger scatter than that for relatively fast cooling rates. The cooling rate parameter needs to be considered in greater detail, and hence more data for relatively slow cooling rates will be collected in 2006.

3.2. Mechanical property test

3.2.1. Ring compression test for Zry-2 cladding (BWR 50 GW·d/t type)

The results of the ring compression test were evaluated from the total crosshead displacement ratio defined in Eq. (3). The total crosshead

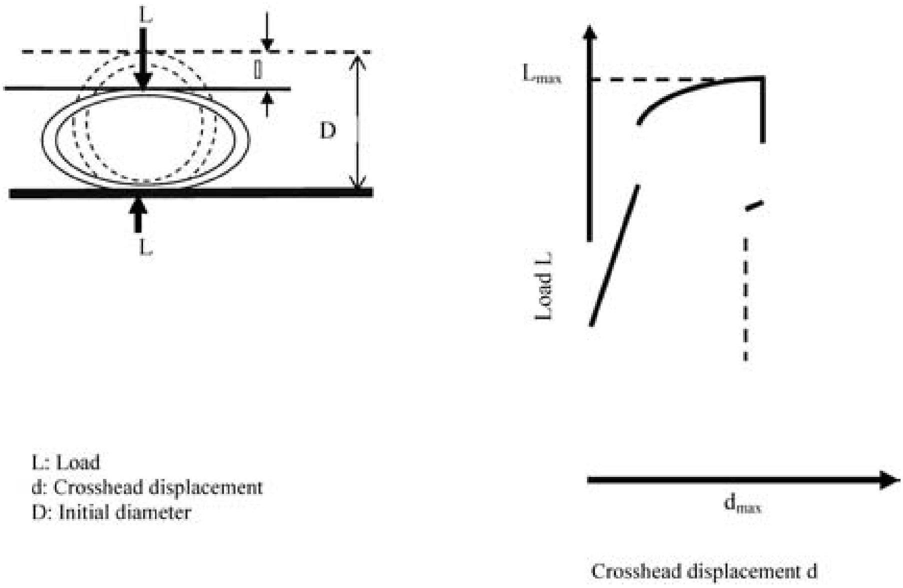

FIG. 7. Measured properties in ring compression test.

displacement ratio can be considered as a ductility index and reflects the total deformation of a ring specimen, including both the elastic and plastic deformation as explained in Fig. 7.

$$\text{Total crosshead displacement ratio } (\%) = \frac{d_{\max}}{D} \times 100 \tag{3}$$

where,

d_{max} = Total crosshead displacement at the first drop of the load in the compression test

D = Initial diameter of the ring specimen.

The results of the ring compression test for Zry-2 cladding are summarized in Fig. 8. The total crosshead displacement ratio does not show a particular hoop stress dependence for the specimens HRT treated at 300°C and 250°C. On the other hand, the total crosshead displacement ratio increased at HRT 0 MPa (only heat treatment) at 400°C compared to 'as irradiated material'. Fig. 9 shows the correlation between the total crosshead displacement ratio and Fe(40°). From Figs 8 and 9, the ductility of Zry-2 cladding can be summarized as follows. At the HRT temperature of 300°C and 250°C, hydride reorientation

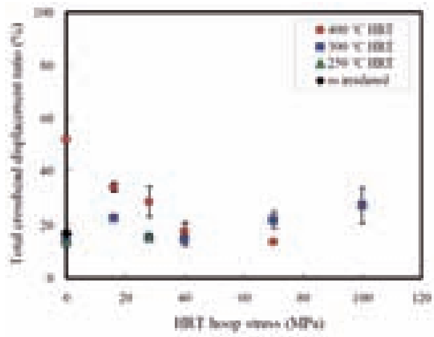

FIG. 8. Correlation between total crosshead displacement raio and hoop stress in HRT for irradiated BWR Zry-2 cladding (cooling rate: 30°C/h).

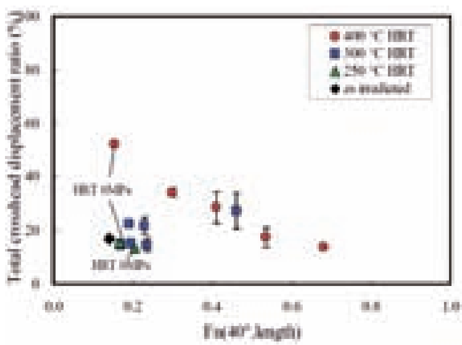

FIG. 9. Correlation between total crosshead displacement raio and Fn(40°, length) for irradiated BWR Zry-2 cladding (cooling rate: 30°C/h).

to radial direction is not very large below the HRT 70 MPa, so that a significant effect of hoop stress on ductility was not observed. The results for specimens treated at HRT 100 MPa at 300°C seem to indicate that the ductility is not affected by large radial hydride orientation. However, this may be better confirmed from data obtained in 2006. At 400°C, the ductility increased compared to 'as irradiated material' by 400°C heat treatment at HRT hoop stress of 0 MPa, and it decreased with the increase of hoop stress in HRT, or the amount of radial hydride. The ductility increase at a HRT hoop stress of 0 MPa at 400°C was surmised to be due to the irradiation hardening recovery of the cladding at HRT temperature.

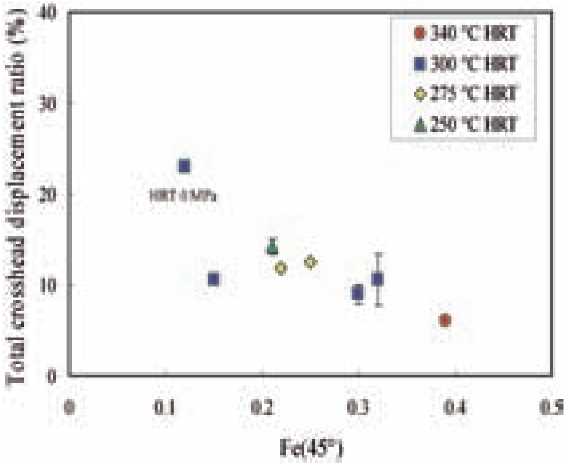

FIG. 10. Correlation between total crosshead displacement ratio and Fe(45°) for irradiated PWR Zry-4 cladding (cooling rate: 30°C/h).

3.2.2. Ring compression test for Zry-4 cladding (PWR 39 and 48 GW·d/t type)

The results of the ring compression test for Zry-4 cladding are summarized in Fig. 10. The total crosshead displacement ratio decreased slightly with the increase of Fe(45°). The specimen at HRT hoop stress 0 MPa at 300°C showed the highest ductility. A threshold in Fe(45°) for the ductility decrease was not observed. The effect of cooling rate on the ductility of cladding is shown in Fig. 11. The total crosshead displacement ratio at 0.6°C/h cooling rate is at the same level as that for a 3°C/h cooling rate, while it increased for the 30°C/h cooling rate.

3.2.3. Longitudinal tensile test

Longitudinal tensile tests for both BWR 50 GW·d/t type Zry-2 and PWR 48 GW·d/t type Zry-4 cladding tube material were also performed (Table 3). Hydrides in the longitudinal cross-section precipitated along the longitudinal axis for each specimen tested after HRT. The effects of HRT on longitudinal mechanical properties, such as yield stress or elongation, were not observed for Zry-2 or Zry-4 cladding materials.

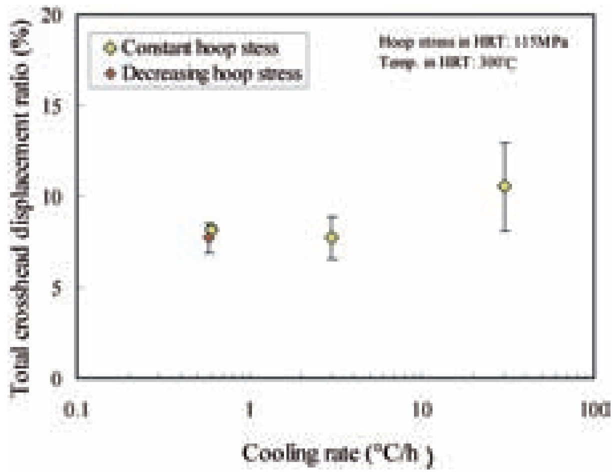

FIG. 11. Effect of cooling rate on crosshead displacement raio for irradiated PWR Zry-4 cladding.

3.2.4. Discussion

The effect of radial hydride on circumferential ductility can be compared between BWR Zry-2 and PWR Zry-4 cladding in Figs 9 and 10 because Fn(40°, length) can be considered to be close to Fe(45°) in the mid-wall area. BWR Zry-2 and PWR Zry-4 cladding tubes showed somewhat different behaviour when comparing circumferential ductility and hydride orientation. The BWR Zry-2 cladding tube has the Zr liner at its inner surface. It is suggested that the Zr liner relaxes the local stress at the inner surface in ring compression deformation and this may contribute to the ring compression behaviour of BWR Zry-2 cladding specimens.

In Figs 8–11, the total crosshead displacement ratio shows a change of over 6% even for the ductility decreased cladding specimens due to high hoop stress or slow cooling rate. However, it should be noted that the measured value was total crosshead displacement, not the real strain or deformation of the specimen. In 2006, additional data from tests, including ring compression tests under relatively high strain speed conditions, will be obtained in order to improve understanding of the effects of hydride reorientation on the mechanical properties of cladding. The test results obtained before 2006 will provide a technical database for fuel integrity evaluation.

4. CONCLUSIONS

Tests to evaluate fuel integrity in interim dry storage carried out by the Japan Nuclear Energy Safety Organization (JNES) comprised: hydride effect evaluation testing, irradiation hardening recovery testing, and creep testing.

The results of hydride effect evaluation testing obtained from 2004 to 2005 revealed the correlation between radial hydride reorientation and HRT conditions, and the correlation between the degree of radial hydride reorientation and the mechanical properties for both BWR and PWR irradiated cladding tubes. In 2006, additional data, including radial hydride reorientation at relatively slow cooling rate HRT or mechanical properties under relatively high strain speeds, will be acquired. The results obtained before 2006 will provide a technical data base for fuel integrity evaluation.

ACKNOWLEDGEMENTS

The authors are grateful to the following. Fuel rods for testing were provided by the Tokyo Electric Power Co. and the Kansai Electric Power Co. Considerable cooperation was obtained in the implementation of PIEs from GNF-J, NFD, MHI, NDC, NFI, and JAEA. We also appreciate the advice of S. Iwata and the members of Subcommittee of Long term Fuel Safety.

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DISCUSSION

R. EINZIGER (United States of America): What was the initial hydride content of the cladding?

M. AOMI (Japan): In the case of the PWR fuel cladding tubes, the planned hydrogen content was less than the terminal solid solubility at hydride reorientation treatment (HRT) temperature. In the case of the BWR fuel cladding tubes, the hydrogen content was from about 100 ppm to about 200 ppm.

Summary Panel 2

DISCUSSIONS WITH CHAIRPERSONS OF SESSIONS 4–6

The following discussion was preceded by short presentations by each of the session Chairpersons. The presentations were preliminary versions of the summaries given in the final session of the conference (Session 8). They allowed the conference participants to make comments on or ask questions about the Chairpersons' views of their sessions.

Chairperson:	**J. Bouchard** (France)
Members:	**T. Saegusa** (Japan) — Chairperson of Session 4 **N. Tikhonov** (Russian Federation) — Chairperson of Session 5 **W. Goll** (Germany) — Chairperson of Session 6

S.K. JAIN (India): I have a question for Mr. Saegusa. The dry storage casks being used in various countries are considered to be good for about 100 years. Does that not mean that there is no urgency about research and development work on deep geological repositories?

T. SAEGUSA (Japan — Chairperson of Session 4): I do not think that we should postpone research and development work on deep geological repositories — the generations that benefit from nuclear power should do all they can to minimize the radioactive waste management problems passed on to later generations.

J. BOUCHARD (France — President): I have an observation and a question on the report of Mr. Goll. As regards the matter of fuel integrity, the definition of 'fuel damage' may be an important topic for international discussion in the future.

Also, the increase in the time for which spent fuel will be stored has implications for the integrity of the fuel. We have seen that the technology of the facilities and the technology of the containers can cope with longer storage times. What about the fuel? Are we completely confident about the behaviour of the fuel over the long term?

W. GOLL (Germany — Chairperson of Session 6): In my view, creep experiments and hydride reorientation analysis are not really aimed at finding out whether the fuel will be able to withstand many decades of storage. They are focused on the initial years of storage, when creep is significant, the temperature is high and reorientation starts. After 5–10 years substantial changes in the material properties are unlikely. So these tests are more or less intended for assessing the short term behaviour of the fuel. For the really long term, there are at present no tests available.

J. BOUCHARD (France — President): Without the necessary experience of the long term behaviour of fuel, licensing for much longer periods becomes a big problem.

R. EINZIGER (United States of America): When the database was put together for determining what the expected performance of the fuel was, the focus was on 100 year storage to start with — to determine what temperature limits and stress limits were needed for 100 year storage. There is nothing particularly sacred technically about 20 year storage. In fact, the longer you store (at least in the case of LWR spent fuel), the more benign the conditions become because the stresses on the fuel decrease. What you do have to worry about, however, is MOX fuel because, as it ages, it generates more gases and the stresses increase, and that needs to be evaluated. Also, one has to remember that the conclusions which we are drawing now relate to LWR spent fuel. They do not apply to aluminium clad fuel. We do not know whether they apply to high burnup fuel because the database was developed for low burnup fuel. There is not much good experience with high burnup fuel. Also, we do not know whether the same conclusions apply to the fuels that have the newer claddings and the newer designs. The database from which we can draw lifetime conclusions is for LWR fuel below 45 GW·d/t with zircaloy cladding. Everything else is uncertain as regards the database.

J. BOUCHARD (France — President): That means that we should encourage research and development in this field because storing spent fuel for longer periods is clearly a current trend.

R. EINZIGER (United States of America): There are definitely some areas where we need more information in order to solidify our position and reduce the uncertainty in our projections.

S. WHITTINGHAM (United Kingdom): In my view, what we are really saying is that it is very important to state what the limitations of our knowledge are at this stage. It would then provide a reference point to revisit over the next decades — particularly when we talk about MOX fuel and high burnup fuel. We could also consider the possibility of entering into joint ventures on research and development in this area.

W. GOLL (Germany — Chairperson of Session 6): I fully agree that the measurements we can perform at present are more or less restricted to the analysis of short term behaviour. The behaviour of — for example — MOX fuel in the long term, with the buildup of helium and the stresses involved, and the behaviour during transport have not been very much addressed by research and development.

LOOKING TO THE FUTURE

(Session 7)

Chairperson

A. MARVY
France

SPENT FUEL MANAGEMENT: PRESENT AND FUTURE

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Abstract

Since the last IAEA conference on spent fuel storage in 2003 there has been a marked change in the environment of the nuclear energy industry in terms of energy security, global warming, waste management, etc. These changes were reflected in the presentations and discussions at the conference. The paper gives an overview of the papers presented at the conference and presents a view of the future in this subject area.

1. INTRODUCTION

The last conference on Storage of Spent Fuel from Power Reactors was held in Vienna, from 2 to 6 June 2003. It was organized by the IAEA in cooperation with the OECD Nuclear Energy Agency. Since 2003, there has been a marked change in the environment of nuclear energy industry in terms of energy security, global warming, waste management, etc. Under such circumstances, the International Conference on Management of Spent Fuel from Nuclear Power Reactors was held in Vienna from 19 to 23 June 2006.

2. SPENT FUEL DISCHARGE

Long term estimates of annual spent fuel discharge were calculated by VISTA [1]. The annual amounts of spent fuel discharged from nuclear power plants and stored worldwide were estimated using three different nuclear power growth scenarios and an assumption of a constant reprocessing ratio

TABLE 1. PARAMETERS FOR PREDICTING NUCLEAR POWER CAPACITY GROWTH

Nuclear capacity growth	Year 2000	Year 2050
P2: High		730 GW(e)
P1: Middle	353 GW(e)	565GW(e)
P0: Low		400 GW(e)

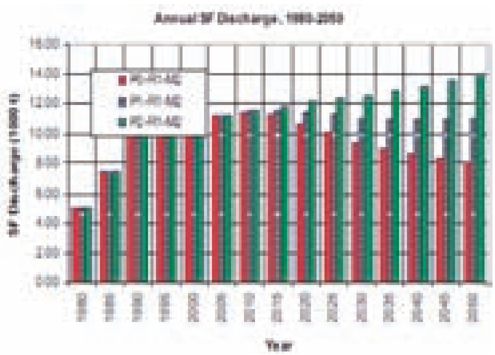

FIG. 1. Annual spent fuel discharge.

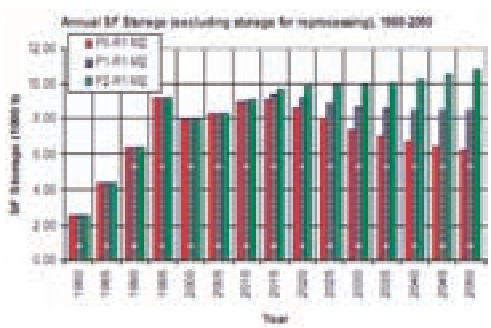

FIG. 2. Annual spent fuel storage.

(30%, designated as R1) and of a Medium MOX refueling ratio (designated as M2). The results are illustrated in Figs 1, 2. Table 1 shows the assumed parameters on nuclear power capacity growth in Figs 1, 2.

The spent fuel discharges are partly suppressed by the increasing average burnup in the reactor mix. Storage technologies are able to cope with the storage requirements. The storage designs and operational specifications can be adapted to the increasing discharge burnups in due time. The large accumulation of stored spent fuel and the multiplication of away from reactor (AFR) storage sites might become a public acceptance problem and a proliferation-resistance concern.

3. SPENT FUEL MANAGEMENT: THE EVOLVING INTERNATIONAL SCENE

3.1. Renaissance of nuclear power

A major difference since the last conference is that the international community is now demanding more nuclear energy. This is partly because the fossil fuel market has become less able to meet the energy demand in the world. The International Energy Agency estimates the demand for primary energy in 2030 to be 59% higher than in 2002. The recent price increase of crude oil further boosts the energy security merits that nuclear energy possesses.

The strong energy demand in Asia has been met by fossil fuels such as coal, natural gas, etc. The increase in energy consumption, which is largely being supplied by imports from other regions, has caused further strains in the energy markets. In addition, the consumption of fossil fuel is required to be constrained in order to prevent global warming. Nuclear energy could meet the energy demand without emitting carbon dioxide and therefore Asian countries have moved to construct nuclear power plants.

In European and American countries the new interest in nuclear power generation is mainly in response to the price increase in crude oil. The enhancement of nuclear energy use is now becoming a core element of energy security considerations. The USA has reopened discussions on the construction of new nuclear power plants. In addition, the USA is to change its spent fuel management policy from direct disposal to reprocessing; a policy strongly proposed in its 2006 Global Nuclear Energy Partnership proposal. In Europe, the temporary halt in the gas supply to Ukraine from the Russian Federation in 2005 triggered a reconsideration of energy generating strategies in favour of nuclear power generation.

The situation is changing at the back end of the nuclear fuel cycle. Reprocessing is seen as beneficial not only because it is an efficient use of energy resources through recycling, but also because of concerns over radioactive waste management.

3.2. Options for spent fuel management

There are currently three options for spent fuel management [2]. These options can be described as:

(1) Reprocessing and recycle of valuable fissile materials ('closed fuel cycle');
(2) Once through cycle ('open fuel cycle');
(3) 'Wait and see' option.

3.2.1. Reprocessing

So far, only 15% of spent fuel has been reprocessed in the world because most countries have selected the once through cycle. During the early years of nuclear power the reprocessing and recycle option was chosen primarily for two reasons. The first reason was that the option was technically available. The second reason was that the residual uranium and plutonium separated from the waste during reprocessing has significant energy value. This energy value could be multiplied many times if fuel is recycled in fast reactors. Recently, another important reason has been discussed, that the toxicity of minor actinides could be reduced if they are burned in fast reactors, thereby reducing the radioactive waste problem.

3.2.2. Once through cycle

There are a number of disparate reasons for choosing the once through cycle. The countries that have an abundance of natural resources have no interest in longer term recycling. Other more general reasons include economic aspects and concern about proliferation issues associated with reprocessing.

3.2.3. 'Wait and see' option

The reasons for adopting a 'wait and see' approach are related to economics, the long timescales involved in securing appropriate disposal facilities, uncertainties about future energy requirements or simply because of a lack of political direction in overall national strategy.

Many countries are following dual strategies, for instance all countries that are committed to a reprocessing and recycle strategy are also committed to a wait and see strategy for a proportion of their fuel discharge.

4. INTERNATIONAL SAFETY REGIME

The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management entered into force on 18 June 2001. The Joint Convention applies to spent fuel and radioactive waste resulting from civilian nuclear reactors and applications and to spent fuel and radioactive waste from military or defence programmes if and when such materials are transferred permanently to and managed within exclusively civilian programmes, or when declared as spent fuel or radioactive waste for the purpose of the Convention by the Contracting Party. The Convention also applies to planned and controlled releases into the environment of liquid or gaseous radioactive materials from regulated nuclear facilities. The Convention calls for review meetings of Contracting Parties. Each Contracting Party is required to submit a national report to each review meeting that addresses measures taken to implement each of the obligations of the Convention.

5. CRITICALITY SAFETY

5.1. Effect of burnup credit

The criticality safety analysis of a transport cask or storage facility has usually been based on the fresh fuel assumption and the subcriticality limit is ensured by providing a conservative safety margin covering all kinds of uncertainties. It was realized that using the planned new advanced fuel types, the presently used transport casks and storage facilities can be used only with reduced capacity in some cases, if burnup credit is not used in subcriticality analysis. These constraints could be removed if the real uranium and plutonium content corresponding to the actual burnup were used.

5.2. Lack of isotopic composition calculation

The largest sources of uncertainties are associated with errors due to nuclear data, the errors in isotopic composition calculations and the uncertainty due to the influence of the axial burnup distribution (end effect). Errors due to the composition calculations can be determined by a comparison of calculations with post-irradiation experimental data. For WWER type fuel only a few such measurements are accessible with well documented irradiation history, and these data have only been available since last year.

5.3. Burnup credit analysis

A reactor pool for the existing fuel types of WWER-440 reactors may lead to violations of nuclear safety criteria under an optimum air–water density of 0.25 g/cm^3. In the analysis with simultaneous credit for boric acid (C_B > 1400 ppm) and fuel burnup the system reaches Keff < 0.95 at the level of 50 MW·d/kg.

6. STORAGE TERM LIMITS

6.1. Life management programme

Upon closure of a nuclear power plant, one option is to transfer the spent fuel from wet to dry storage. A life management programme for long term operation of spent fuel storage is extremely important. Application of an ageing management methodology to spent fuel storage includes the screening of the installations and determination of the critical systems, structures and components and the determination of the principal ageing mechanisms and stressors.

6.2. Long term spent fuel management for high burnup fuel

Geological disposal as a long term spent fuel management strategy has been developed. A longer cycle operation of nuclear reactors corresponds to higher burnup of spent fuel and a resulting increase in radiation levels, decay heat and radionuclide inventory. These impacts on the safety of geological disposal and the long term management of spent fuel have to be considered.

7. STORAGE FACILITIES

The following is a summary of some of the papers presented in this session of the conference.

(a) Metal cask for transport and storage in the Czech Republic. At the Dukovany nuclear power plant 60 metal casks for transport and storage of of a capacity of 600 t HM have been used since 1995. Additionally, a modified metal cask for transport and storage was licensed in 2005. At the Temelin nuclear power plant a similar cask storage facility with a capacity of 1370 t HM will be in operation in 2014. A backup site in Skalka may serve as an underground cask storage facility with a horizontal access shaft. Contemporaneously with the preparations for the deep repository, the possibilities of recycling and new technologies aimed at decreasing the volume and toxicity of the spent fuel will be pursued. A deep repository is planned to be put into operation by 2065.

(b) Horizontal concrete silo in Argentina. A modular storage facility has been designed to store spent fuel from the Atucha-I PHWR power plant. The facility is composed of concrete structures and metallic canisters, each of which contains 37 spent fuel assemblies and is backfilled with helium.

(c) Safety of dry storage containers in Canada. Dry storage container systems are operated by Ontario Power Generation. The radiation dose consequences to the public and workers of postulated credible accidents were assessed. The worst case scenario assumed 30% fuel failure within a dry storage container, as well as failure of the container transfer clamp seal, allowing the release of 30% of the Kr-85 and H-3. The consequences to the public from this release were found to be less than 0.2% of the regulatory limit.

(d) Horizontal concrete silo in Armenia. Since 1996 in Armenia 616 spent fuel assemblies of the WWER type have been stored using NUHOMSÆ storage modules and additional storage capacity of NUHOMSÆ is now being planned.

(e) Construction of the first spent fuel storage facility in the Russian Federation. Construction of the first storage facility for spent fuel from WWER-1000 and RBMK-1000 reactors began in 2004 in the Russian Federation. The whole projected capacity of the storage facility is more than 33 000 t U. The storage system has two physical barriers, hermetic containers and hermetic storage tubes. Requirements for spent fuel storage are that the storage duration is not less than 50 years, the storage medium is nitrogen, the storage temperature is from 300 × C to 350 × C , and water content is less than 25 g/cm^3.

(f) Licensing and inspection experiences in the USA. The US Nuclear Regulatory Commission (NRC), through the combination of a rigorous licensing and inspection programme, ensures the safety and security of dry cask storage. The first licence was issued to the Surry Nuclear power Plant in 1986 Today there are over 30 independent spent fuel storage installations (ISFSI) currently licensed by the NRC with over 700 loaded dry casks and there will be 50 ISFSIs by the year 2010. No release of spent fuel dry storage cask contents or other significant safety problems from the storage systems in use today have been reported.

(g) Spent fuel storage in Bulgaria, Finland, Slovakia and Ukraine. In Bulgaria, a wet storage facility (four pools) was built and commissioned at the Kozloduy nuclear power plant in 1990 for spent fuel of of the WWER-440 and WWER-1000 types. A dry storage facility (casks) has been planned for construction at the Kozloduy nuclear power plant. In Finland, capacity extension of pool storage facilities has been planned based on a shutdown plan of the plants and spent fuel disposal in 2010. In Slovakia, discharged spent fuel has been stored in an interim spent fuel storage pool. In Ukraine, spent fuel from WWER-1000 reactors has been stored in concrete casks at an interim AFR spent fuel storage facility at the Zaporizhya nuclear power plant. The ISFS design was significantly updated by the development of a methodology for the practical implementation of burnup credit.

(h) Spent fuel from research reactors in Egypt and Indonesia. In Egypt, defects appearing in fuel elements motivated an investigation of the reactor fuel and the design of a new system of encapsulation of the defective fuel elements for long term storage. Fuel is stored in a stainless steel tank filled with water and located in a concrete pit below ground level. The capacity of the storage tank is up to 176 fuel assemblies. The defective fuel assembly is encapsulated in a tightly closed aluminium tube filled with nitrogen containing 5% of helium as an inert gas to prevent the corrosion of fuel rods. Indonesia has produced spent fuel from three research reactors and a licensing framework for a spent fuel storage installation was discussed.

8. STORAGE CONTAINERS

The following is a summary of some of the papers presented in this session of the conference.

(a) Metal cask for transport and storage. At the Dukovany nuclear power plant in the Czech Republic a modified CASTOR©440/84M cask was licensed in 2005, with improved neutron shielding properties, modified trunnion construction, optional use of a third welded lid, and a new fuel basket design. The cask can be loaded in three basic configurations of thermal output of fuel assemblies, homogeneously and heterogeneously.

(b) Research and development for fuel with higher enrichment, burnup, heat load and radiation level. Cogema Logistics has been implementing a research and development programme focusing on new basket designs and materials with high boron content for sub-criticality, high performance thermal transfer systems, new neutron shielding materials resistant to severe conditions, and new shock absorbing covers to keep the G loads low.

(c) Demonstration tests using full scale metal cask and concrete cask. For concrete casks, the Japanese Central Research Institute of Electric Power Industry (CRIEPI) reported heat removal tests using full scale casks and verified the applicability of a thermo-hydraulic analysis method. Drop tests using full scale canisters were also executed.

(d) Metal and concrete cask for transport and storage. A complex metal and concrete structure design in the form of three layers of special steel with reinforced concrete filling between them has been developed in the Russian Federation. The choice was made in view of their lower price and greater capacity. Eighty-six casks had been produced by April 2006 and the production is increasing.

(e) Transport cask for spent fuel for research reactor. A metal transport cask design was reported for spent fuel of Russian origin for a research reactor LVR-15 in the Czech Republic. The cask takes 36 assemblies with a thermal power of 450 W.

9. FUEL AND CLADDING PROPERTIES AND BEHAVIOUR

The following is a summary of some of the papers presented in this session of the conference:

(a) Damage in spent fuel. The NRC's ISG-1 and the American National Standards Institute's (ANSI) guidance do not provide the logic behind their definitions of damaged fuel. The rationale for the definition was given in a paper. Essentially, damage is defined by the requirements of the system and those of the regulations, and fuel may be considered damaged under one scenario but undamaged under another.

Management criteria for damaged spent fuel in long term storage were discussed. They included classification of defects, identification of defects in storage, the design of facilities for damaged spent nuclear fuel, retrieval, packing, selection of container types for long term storage, and ensuring safety during transport and storage.

(b) Trends in dry spent fuel storage and transport in Germany and the USA. In Germany, the present engineering approach to ensure cladding integrity is to impose limits of 1% plastic strain and 120 MPa tangential (hoop) stress. These values limit thermal creep degradation and hydride reorientation under dry storage conditions.

In the USA the trend to higher burnups has required extension of the database for the cladding integrity assessment. At the beginning of dry storage, where temperatures of up to about 400°C can be reached, hydrogen is dissolved and will only be precipitated at lower temperatures and, hence, lower stresses. As a result, hydrogen reorientation will largely be avoided.

(c) Research and development on spent fuel integrity and characteristics. The Japan Nuclear Energy Safety Organization (JNES) reported on its test programme on hydride effect evaluation testing, irradiation hardening recovery testing and creep testing. The results of hydride reorientation testing clarified that the degree of hydride reorientation depends on the hoop stress and temperature.

Spent fuel characteristics for RBMK type reactors are almost absent in the scientific literature. Modelling of the RBMK-1500 spent nuclear fuel characteristics and comparison with available experimental data were reported.

10. LOOKING TO THE FUTURE

Recently, international management initiatives were presented for the purpose of preventing proliferation and improving world security while maintaining the ‘inalienable rights’ of peaceful uses of nuclear energy as set out in the Non-Proliferation Treaty, as shown in Table 2 [3–5]. Although foci and priorities may differ, common goals are shared between these proposals.

In 2006, the US Department of Energy proposed “The Global Nuclear Energy Partnership” [7]. It intends to be a comprehensive strategy to increase US and global energy security, to encourage clean development around the world, to reduce the risk of nuclear proliferation, and to improve the environment. The USA will work with other advanced nuclear nations to develop a fuel services programme that would provide nuclear fuel and recycling services to nations in return for their commitment to refrain from developing enrichment and recycling technologies.

The role of spent fuel storage in multinational approaches to the back end of the fuel cycle was discussed. In the SAPIERR project, organizations from 14 European countries with interests in the concept of shared repositories

TABLE 2. THE AIMS AND COMPONENTS OF THE INTERNATIONAL MANAGEMENT INITIATIVES ON NUCLEAR MATERIALS AND TECHNOLOGIES

			'El- Baradei Plan'	IAEA Experts' Group report	Bush Administration
Title upon dissemination			curbing the nuclear threat, Feb. 2005.	Multilateral approaches to the nuclear fuel cycle, Feb. 2005.	Strengthening international efforts against WMD proliferation, Feb. 2004.
States or entities targeted	Non- NPT states or uncoupling member states within NPT regime	1. Deterrence against new acquisition of nuclear weapons and/ or withdrawal from NPT regime.	Strengthened functioning of UN Security Council. — call on the UN Security Council to act swiftly and decisively in the case of any country that withdraws from the NPT	Exploring opportunities for multi-National approaches (MNA). — Promoting voluntary conversion of existing facilities to MNAs, and pursuing them as confidence building measures, — Creating, through voluntary agreements and contracts, multinational, and in particular regional, MNAs for new facilities.	Strengthening the International Atomic Energy Agency (IAEA). — all states should sign the IAEA Additional Protocol, — the IAEA Board of Governors should create a special committee on safeguards and verification, — no state under investigation for proliferation violations should be allowed to serve or continue serving on the IAEA Board of Governors or on the new special committee.
	Complying and non-weapon (peaceful user) states.	2. Severer obligations and limitations to rights for peaceful uses, to help deterrence against new acquisition by other malicious states or entities.	Enhanced inspection and regulation on illicit trades. — urge states to act on the security Council's recent resolution 1540, — raise the bar for inspection standards by establishing the 'additional protocol' as the norm for verifying compliance with the		

TABLE 2. THE AIMS AND COMPONENTS OF THE INTERNATIONAL MANAGEMENT INITIATIVES ON NUCLEAR MATERIALS AND TECHNOLOGIES (cont.)

			'El- Baradei Plan'	IAEA Experts' Group report	Bush Administration
Title upon dissemination			curbing the nuclear threat, Feb. 2005.	Multilateral approaches to the nuclear fuel cycle, Feb. 2005.	Strengthening international efforts against WMD proliferation, Feb. 2004.
States or entities targeted	Complying and non-weapon (peaceful user) states.	Moratorium on construction and operation of sensitive facilities.	Moratorium for new sensitive facilities, and fuel supply guarantee in exchange of them. — put a five-year hold on additional facilities for uranium enrichment and plutonium separation. — commit the countries that already have the facilities to guarantee an economic supply of nuclear fuel for bona fide uses.		Controls against enrichment and reprocessing. — the members of the Nuclear Suppliers Group ensure that states which renounce enrichment and reprocessing technologies have reliable access, at reasonable cost, to fuel for civilian reactors. — the Nuclear Suppliers Group should refuse to sell uranium enrichment or reprocessing equipment or technology to any state that does not already possess full-scale, functioning enrichment or reprocessing plants.
		3. Guaranteed supply of nuclear fuel materials and services for peaceful uses, to compensate those obligations and limitations implemented.		— Reinforcing existing commercial market mechanisms, — Developing and implementing international supply guarantees with IAEA participation.	
	Nuclear weapons states (NWSs).	Accelerated efforts for nuclear disarmament.	— call on the five nuclear weapon states party to the NPT to accelerate implementation of their 'unequivocal commitment' to nuclear disarmament.		

collaborated to define an inventory for such a facility. The options for implementation (not yet including identification of potential sites) were examined and a first consideration of the influence on storage and transport were discussed.

The feasibility of plutonium use in BWR reactors as a way to dispose of spent fuel was discussed by Instituto Nacional de Investigacious Nucleares, Mexico. It was concluded that even with the current higher costs of uranium, the recycling option is more expensive than the once through option. However, the reduction of high level waste will be approximately 68%, which is significant.

11. CONCLUSION

Spent fuel management in the world is continuously evolving, reflecting energy security, environmental issues such as global warming, radioactive waste management, non-proliferation, public acceptance, etc. Exchange of information and data on spent fuel storage technologies and public acceptance matters should be continued. It is important to collaborate internationally on spent fuel management.

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DISCUSSION

A. MARVY (France — Chairperson): The knowledge that we have accumulated about spent fuel is impressive, but there is a need to increase it in order to meet future goals and the possible expansion of nuclear activities in the world.

Yesterday, I think someone said that the nuclear community should also state where the limits of our knowledge lie in order to possibly increase public acceptance. Perhaps that is a point which we should bear in mind when carrying out research or working as engineers, thinking about how we might include the public acceptance issue in that context.

I should like to know what Mr. Saegusa's views are regarding the public acceptance issue, as I understand that Japan is deeply involved in trying to make nuclear activities more comprehensible and to share the knowledge acquired with the public at large.

T. SAEGUSA (Japan): In my country, the question of the duration of interim storage has been discussed with the public, particularly at the local community level, and a maximum of 50 years has been decided on as a result. Even so, the regulatory body will issue a licence for just a few years, the licence having to be renewed every so often after periodic inspections. At all events, the consulted communities did not want to host a disposal site.

We need more away-from-reactor sites for spent fuel storage, and several candidate sites are currently the subject of discussions between the utilities and the public. The results are due to be announced in the near future.

C. GOETZ (Germany): The fact that a tunnel facility for the storage of spent fuel has been licensed in Germany does not reflect a new philosophy. At the same time, over ten surface facilities have been licensed. The tunnel facility licence was issued simply because at the site in question there was no room for a surface facility.

T. SAEGUSA (Japan): In my country there are relatively few flat areas and the utilities may prefer digging a tunnel into a hill rather than levelling the hill.

J. WHANG (Republic of Korea): Mr. Saegusa spoke about a full scale metal cask demonstration test being carried out in Japan for public acceptance reasons. Who raised the public acceptance issue and how was it resolved as a result of the test?

T. SAEGUSA (Japan): There were no specific comments from the public about the test, but in the light of our experience with transport casks we decided to carry out a full scale test for storage containers. The public has always had concerns about technologies which are discussed exclusively among experts, and a full scale test is easily understood not only by experts but also by the public.

A. MARVY (France — Chairperson): After 15 years of research and development work in France on various options for radioactive waste management, we came to the conclusion that ‘hands-on’ demonstrations of what is usually kept invisible or discussed behind closed doors can be very helpful in dealing with the public and with politicians.

CONCLUSIONS

(Session 8)

Chairperson

J. BOUCHARD
France

SUMMARY OF SESSION 1.A

SPENT FUEL MANAGEMENT — THE EVOLVING SCENE*

Chairperson

J. BOUCHARD
France

The session consisted of four presentations on national policies and three on international perspectives. The four national presentations showed how the nuclear energy scene is changing.

The US initiative, the Global Nuclear Energy Partnership (GNEP), aims to facilitate a substantial increase of the use of nuclear energy in the world while reducing the radioactive waste and minimizing proliferation risks. Three main elements can be identified in this new US initiative:

(a) Recognition of the need for an increasing use of nuclear energy in the world;
(b) The recognition that only an advanced closed fuel cycle can sustain such a development while solving waste management and non-proliferation issues;
(c) The offer of a partnership to countries willing to develop a nuclear energy capacity.

The Russian initiative also recognises the need to increase the use of nuclear energy and includes an offer to provide global nuclear services to countries which do not have their own fuel cycle infrastructures. Besides expert training and offers to build reactors, the Russian Federation is proposing to provide fresh fuel with the spent fuel coming back to the Russian Federation and/or possibilities of interim storage of spent fuel prior to reprocessing in a closed fuel cycle.

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The third national presentation was on the spent fuel management policy in France. The country remains committed to its nuclear energy programme and, in addition, to the decision to build a European pressurised water reactor (EPR), a Gen III reactor, and to prepare a prototype of the Gen IV system. Recently, Parliament approved a Waste Management Act. The main features of this new law are the following:

(1) Reaffirmation of reprocessing and recycling as the national policy for spent fuel management, with a continuous effort to improve the closed fuel cycle;
(2) A commitment to open a geological repository for high level waste by 2025;
(3) A clarification that interim storage will be used only to provide flexibility before reprocessing operations and/or waste disposal.

As a fourth illustration of how the nuclear energy systems in the world are changing rapidly there was a presentation of the nuclear energy prospects in India. There are very great needs for energy in this country, and part of them will be provided for by nuclear power plants. The development of a closed fuel cycle with fast reactors and the progressive use of thorium are important features of nuclear energy policy in India.

From these four national presentations it is possible to identify some important trends related to spent fuel management:

(i) Spent fuel is to be reprocessed, sooner or later;
(ii) There will be an increase of spent fuel transport from country to country;
(iii) Storage of spent fuel is only considered as an interim buffer to allow flexibility for reprocessing and waste disposal.

The IAEA and OECD/NEA presentations recalled that the international scene in spent fuel management is still varied, with some countries intending to dispose of the spent fuel in geological repositories, following the once through policy, while others are still considering the two options and using storage as a tool in a kind of ‘wait and see’ policy.

The multinational approach for the management of fuel cycle facilities, including waste disposal at a regional repository, was discussed following the presentation of the work done by the expert group set up by IAEA Director General M. ElBaradei. Nuclear security issues were also presented and discussed.

DISCUSSION

S. WHITTINGHAM (United Kingdom): To someone who, like me, does not belong to the spent fuel management community there seems to be a view here — perhaps based on the excellent safety record of radioactive material transport — that the transport of spent fuel will simply 'happen'. It should be borne in mind, however, that the excellent safety record is due to the existence of a set of international regulations and a set of international standards and to the fact that the entire transport community works on those regulations and standards. As I said in my presentation, in the field of radioactive material transport there is a true safety culture.

I hope that the spent fuel management community will ultimately adopt international standards. If it does not, the transport community will have to pick up the pieces, and the spent fuel management community may well find that some of the spent fuel which it is storing cannot be transported in the public domain.

Imagine the following likely future problem — spent fuel loaded into a storage cask 70 years previously has to be transported in the public domain from a storage pond to a repository. What questions will people ask themselves regarding the safety of that spent fuel? I hope they will not be saying "If only our ancestors had thought of this problem!" It is unlikely that anyone will want to open the cask, so it will have to be transported as it is. Consequently, there will be a need for international safety standards that have evolved over the decades so that people know what they are dealing with. The transport community has such standards, which explains its success in the area of radioactive material transport.

SUMMARY OF SESSION 1.B

INTERNATIONAL SAFETY REGIME*

Chairperson

A. GONZÁLEZ
Argentina

1. INTRODUCTION

The Conference discussed the main elements of the international nuclear safety regime that is being built under the aegis of the IAEA. It includes legally binding undertakings in the form of international conventions, international safety standards, and international provisions for the application of these standards. In relation to spent fuel management, the Conference particularly addressed the implementation of the obligations undertaken under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (the Joint Convention) and the establishment of relevant safety standards. The Conference was informed about the successful experience in establishing an international safety regime in the area of the safe transport of radioactive material. This type of regime provides for a global harmonization of safety and is an essential element for facilitating trade among countries.

2. JOINT CONVENTION

The Conference was informed that the Joint Convention is the major international legally binding instrument for the safe of management of spent fuel. The Contracting Parties to the Joint Convention are responsible for implementing their obligations under the convention and the IAEA serves as a

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managing secretariat. The second review meeting of the Joint Convention was held in May 2006, and the results of this meeting were presented to the Conference.

It was noted that the reporting and review mechanisms under the Joint Convention are under discussion by the Parties but that, as yet, there is no common view on possible modifications or the strengthening of the mechanisms. Maintaining or improving the rigour of this process and avoiding that bad practices are hidden or are not identified will be a real challenge for the Contracting Parties in the future.

A similarly important point concerned the potential application of international safety standards within the review process of the Joint Convention. The preamble of the Joint Convention refers to 'international safety standards'. The application of international safety standards in the context of the Joint Convention may improve the reporting and reviewing by making the process more harmonized and comparable. However, for the present, their use in this context is left to the discretion of each of the Contracting Parties.

The Conference noted that the Joint Convention is an 'incentive' convention, basically limited to a review of safety among the Contracting Parties. The contrast between 'incentive' versus conventions with sanctions was noted. It was mentioned that reassurance of compliance with the support of an international safety regime could improve the rigour of the reporting and review process. However, at present the Contracting Parties are not in a position to go in that direction, inter alia, because the elements of an 'international safety regime' and their application in this subject area are not yet well enough or precisely defined so as to be applied in an international legal instrument.

Membership of the Joint Convention was another issue brought to the attention of the Conference. Most of the countries operating nuclear power plants are Contracting Parties to the Joint Convention, but some important 'nuclear' countries are not yet Parties to it. All countries with operating nuclear reactors (either power or research reactors) generate spent fuel that needs to be managed in a safe manner. Moreover, all countries around the world generate radioactive waste, whether from the nuclear industry, or from medical applications, or from other industries such as the oil industry. It is obvious, therefore, that the Joint Convention should be of interest to all countries and not only to nuclear power countries.

3. RECOMMENDATIONS

(a) States should be encouraged to become Parties to the Joint Convention in order to report on the safety of their management of spent fuel in accordance with the requirements of the Joint Convention.

(b) The IAEA should enhance its efforts to foster information exchange on the Joint Convention, with the aim of increasing the number of Contracting Parties.

(c) The IAEA should encourage the exploration among Contracting Parties of possible mechanisms for strengthening the Joint Convention's reporting and reviewing processes, particularly in relation to the safe management of spent fuel, and report any findings to the Contracting Parties for their consideration.

4. SAFETY STANDARDS

The Conference noted that the IAEA is entitled under its Statute to establish international safety standards and to provide for their application following the request of a State. These standards are mandatory for the IAEA's operations and projects, including technical cooperation projects, and may be used, inter alia, as a reference for national regulations or directly as national regulations or for the harmonization amongst countries of regulatory/legal systems. The Conference was informed that, for instance, such harmonization efforts are under way through the work of the Western European Nuclear Regulators Association (WENRA). Such standards might also be used in the reporting and review process of the Joint Convention (see above) and they could also play an important role in multinational and international approaches and 'partnerships' for the management of spent fuel. A real partnership in this area would need an appropriate safety framework.

The Conference welcomed the news that, finally, the IAEA Commission on Safety Standards has approved a common safety 'fundamentals' document. This should facilitate harmonization among the different areas in which international safety standards are being prepared, namely radiation safety, nuclear safety, waste safety and transport safety. It noted that the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS) is undergoing a process of review. It also welcomed the decision that the current review process of the BSS will include converting the BSS to a very much needed conceptual bridge between the Safety Fundamentals and the facility related safety standards.

The international safety standards on the safe transport of radioactive materials (the IAEA Transport Regulations) were highlighted in this connection as a good example to be followed in the establishment of a wider nuclear safety regime. The Conference was briefed in detail about the progress and experience in establishing international standards for transport. In this context, trends in fuel management strategies were reviewed, including higher discharge burnup and long term storage at the reactor site prior to disposal. These strategies pose technical issues that need to be addressed before the fuel is discharged and stored so that appropriate storage regimes can be put into place to minimize future issues that may arise at the time of transport, which may be some decades in the future.

The exemplary safety record of spent fuel transport was emphasized. This record is due to the high professionalism and technical competence of those involved and the universal acceptance of the IAEA's Transport Regulations. The Conference noted that the involvement of all stakeholders in the continued development of the IAEA Transport Regulations and the international agreements in which the requirements feature is a key component in fostering an effective safety culture.

The Transport Regulations contain a vast amount of detailed quantitative information and provide a solid basis for the execution of the respective requirements. The Conference noted the incompleteness of the suite of international standards on the safety of spent fuel management. To have a solid and complete suite of safety standards is important, inter alia, for facilitating the provision of appraisal services requested by States in accordance with the IAEA Statute.

5. RECOMMENDATIONS

(a) The IAEA should continue to strengthen its efforts to establish a complete system of international safety standards; the good example of the Transport Regulations could be used as a model. Particular efforts should be made to review the existing draft and planned standards on the safety of spent fuel management for completeness.

(b) States are encouraged to make good use of the IAEA Safety Standards.

(c) States might also consider making use of IAEA appraisal services in order to calibrate their national legal/regulatory frameworks and their national practices against international standards.

SUMMARY OF SESSION 2

CRITICALITY SAFETY*

Chairperson

J. WHANG
Republic of Korea

The session consisted of a keynote presentation and three national presentations. The keynote presentation provided information on state of the art burnup credit criticality safety analysis methods. In addition, it provided an overview of national practices, ongoing activities and the regulatory status of the use of burnup credit in different countries. The presentation dealt with all of the important aspects to be considered when criticality safety or burnup credit is to be applied to a spent nuclear fuel system from generation to disposal, including the principles for choosing calculation procedures, the verification and validation of calculation procedures, the impact of conservatism, and reactivity bias due to isotopic bias. In the subsequent discussion two important issues were raised, the application of the double contingency principle for the misloading of spent fuel and decisions on the number of isotopes to be taken into account for burnup credit calculations.

The benefits of, and the need for burnup credit were discussed in a paper from Hungary. Regulations covering criticality safety have recently been introduced and the benefits of applying burnup credit were demonstrated with some sample cases. However, there are questions which need further investigation to ensure the safe application of burnup credit in this case. These include the influence of the spatial distribution of burnup, and errors in cross-section determination and in composition calculations.

Experience of spent fuel pool re-racking and the use of casks to provide contingency storage space was described in a paper from South Africa.

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A paper from Ukraine described criticality calculations for the reactor pool of a WWER-440 reactor. The authors intended to show that the extent of conservatism could be determined and reduced by using the state-of-the art calculation studies.

The overall conclusions that can be drawn from the session are as follows:

(a) The interest in burnup credit is mainly based on the benefits that can be obtained from the capacity extension of existing facilities, which has large financial implications.
(b) The initial enrichment of fresh fuel and the degree of burnup of spent fuel are basic elements for determining the criticality safety of spent fuel from cradle to grave.
(c) Whatever the motivation, criticality safety using burnup credit should not be seen as a straightforward approach, but rather as a complex matter in which sub-criticality has to be ensured under all conditions, including design basis events, and taking into consideration all uncertainties in calculation and in isotopic composition.
(d) In practice, the extent to which credit is taken for burnup can vary. It can range from taking credit for just the net fissile content of the fuel to a full burnup credit which could include actinides, fission products and the burnout of absorbers. If a type of burnup credit is to be used, an appropriate rigorous validation and verification of isotope inventories and of calculation tools is required. The more burnup credit that is claimed, the more proof (validation and verification) is required.
(e) The presentations showed that quite a lot of published information on isotopic composition is available from experimental investigations for western reactor types, while, on the other hand, little published information seems to be available for WWER reactors. This leads to differences in the approaches being adopted in the application of burnup credit and raises the question of what should or could be done to improve the situation, including possible international assistance by the IAEA.
(f) A deeper look into the details of burnup credit, e.g. the dependence of isotopic composition on the hardening of neutron spectrum during reactor operations, on irradiation history and the non-uniformity of burnup, raises the question of how far to go in giving credit for burnup and where to stop?
(g) Finally, the calculation tools used for criticality assessment must be checked to ensure that they are capable of representing actual situations correctly. This subject did not receive very much attention in the session.

DISCUSSION

J.-C. NEUBER (Germany): You raised the question of where to stop — and is there any standard? They are two separate questions. Regarding the first one, usually you have to produce a loading curve for your spent storage system. Any loading curve has to be applicable to any storage position inside the storage system, so you have to generate a bounding irradiation history and a bounding profile, and that answers your question about where does one have to stop. Of course it depends on the spent fuel storage system one is dealing with and it depends on the level of burnup credit you are choosing.

As regards standards, in Germany we have standards for using burnup credit that answer such questions, and, in the USA also, when its standard is worked out, such questions will be answered.

SUMMARY OF SESSION 3

STORAGE TERM LIMITS*

Chairperson

W. BRACH
United States of America

The purpose this session was to discuss the policy, programme requirements, and technical considerations in the establishment of both initial licence time periods (or terms) for spent fuel storage and the terms for subsequent renewal periods. The session included a keynote paper presentation and a five member panel presentation and discussion both amongst the panel members and with conference attendees.

The keynote paper addressed the licensing and technical review issues which should be considered in reviewing and approving a spent fuel storage facility as a function of the timeframes or term limits. Key elements considered were ageing management with monitoring and surveillance programmes to confirm the continued acceptability of dry cask storage system materials and structures. Institutional controls, including management controls were also significant areas that were addressed.

The session then moved to a panel discussion on Term Related Licensing Issues. 'Term related' refers to the time period of concern, such as short, medium, and long term for the initial or relicensed term.

Each panel member provided a brief overview of spent fuel storage issues experienced and/or studied in their respective national programmes. Common themes in the presentations and the subsequent discussion were: (1) effects of ageing on structures and materials; (2) importance of surveillance and monitoring programmes in support of maintenance; (3) security and physical protection (post 9/11 concerns); (4) integrity of spent fuel after long term storage; (5) changes in design codes and requirements; (6) the opportunity to re-examine assumptions and strategies in the safety basis for the facility during

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relicensing; (7) should new facilities be built rather than relicensing existing facilities?; (8) considerations involved in the transition from a production facility to a non-operating, storage only, facility; (9) public involvement/awareness in relicensing.

While the panel was not structured or intended to provide recommendations, the following points were repeatedly stressed during the presentations and dialogue:

(a) The identification of time periods or 'terms' for extended storage of spent fuel is based on the technical evaluation of the structures and facilities and consideration of institutional and public interests (extended storage terms have been approved in some countries for 40, 50 and 60 years);
(b) Increasing amounts of spent fuel are going into dry cask storage as opposed to wet storage;
(c) Spent fuel dry cask storage studies demonstrate that fuel and cask/canister materials in dry cask storage systems maintain their integrity;
(d) There is a need to determine the type of surveillance and maintenance needed for each component that is important for performing its intended function;
(e) Establishing monitoring and maintenance requirements for the long term is a challenge;
(f) Understanding the long term condition of spent fuel in storage and maintaining its integrity are very important and necessary in relation to ensuring the safety of the eventual transport of the fuel to a reprocessing or disposal facility;
(g) There is a need to address new and evolving technical issues such as the long term performance of materials in the dry cask storage system, effects on structures in a coastal environment, and the understanding of seismicity and high burnup fuel characteristics;
(h) Public and local stakeholder engagement is very important for siting and relicensing of storage facilities.

General conclusions

The panel discussions did not identify any specific issues or impediments to the safe storage of spent fuel. The studies to date and the worldwide performance to date demonstrate that spent fuel in both spent fuel pools (wet storage) and dry cask storage systems can be safely and securely stored. The panel discussions did identify that there is more work to be done to address certain technical issues as discussed above, as well as the need to engage and inform the public on considerations for licensing terms for spent fuel storage.

The panel members and conference attendees discussed a number of ongoing national and multinational efforts to address these issues. Continued IAEA involvement in supporting and facilitating the sharing of developments in storage term limits is encouraged.

DISCUSSION

J. NISAR (Pakistan): Since the nuclear power generating countries have vast experience of both the wet and the dry technologies of spent fuel management, I would propose that we move, through the IAEA, to some type of standardization on storage term limits and, if possible, on technologies.

W. BRACH (United States of America — Chairperson of Session 3): That proposal is in line with a number of recommendations that have been made regarding the consideration that should be given by the IAEA to the development of standards and guides for supporting spent fuel management.

A. GONZÁLEZ (Argentina): Session 3 illustrates the need for international harmonization. For example, as you [Mr. Brach] said, we are freely using the expression 'long term storage' although it has not been defined and has different meanings for different people.

W. BRACH (United States of America — Chairperson of Session 3): Regarding the definition of 'long term', I think it should be based on considerations of safety analysis.

SUMMARY OF SESSION 4

STORAGE FACILITIES*

Chairperson

T. SAEGUSA
Japan

In this session, various national experiences with spent fuel storage facilities were described. The session consisted of five national presentations and a summary by a rapporteur of the ten contributed posters on this subject.

The first paper described a modular storage system designed to store the spent fuel from the Argentine Atucha-I PHWR power plant beyond the existing planned lifetime of the power plant. The facility is composed of concrete structures and metallic canisters, each of which is backfilled with helium and contains 37 spent fuel assemblies.

The second paper described a safety assessment of the dry storage container system used by Ontario Power Generation in Canada. The worst case scenario assumed 30% fuel failure within a dry storage container, as well as failure of the container transfer clamp seal allowing the release of 30% of the ^{85}Kr and ^{3}H. The consequences to the public from this release were estimated to be less than 0.2% of the regulatory limit.

The third presentation described the successful application of the NUHOMS storage technology, widely used in North America, to the spent fuel of the Armenian Nuclear Power Plant, a WWER-440. The dry storage system is modular and each module consists of a metallic canister housed inside a horizontal concrete storage module. The system is flexible and more modules can be added as needed.

As yet, no dry storage facilities for spent fuel exist in the Russian Federation and the fourth presentation described the licensing procedures for the first dry storage facility, the XOT2. The main emphasis in the licensing process has been on the safety justification, with particular consideration being

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given to the following issues: influence of an earthquake, efficiency of barrier seals, evaluation of possible gaseous release and management of radioactive waste.

In the fifth presentation, the US Nuclear Regulatory Commission's (NRC) licensing and inspection experiences were presented. Since 1986, the NRC has licensed over 40 independent spent fuel storage installations with over 800 loaded dry casks. Current projections identify over 50 such installations by the year 2010. No releases of spent fuel dry storage cask contents or other significant safety problems from the storage systems in use today have been reported.

The rapporteur for the session summarized ten posters. They included experience in the Czech Republic on the development and use of metal casks for the transport and storage of spent fuel; the construction of the first dry spent fuel storage facility in the Russian Federation; the arrangements for spent fuel storage from nuclear power plants in Bulgaria, Finland, Slovakia and Ukraine and from research reactors in Egypt and Indonesia.

In summary, the session showed good examples of proper design and engineering solutions and the very high level of technical and organizational readiness of the nuclear community to meet the demands of spent fuel storage.

SUMMARY OF SESSION 5

STORAGE CONTAINERS*

Chairperson

N. TIKHONOV
Russian Federation

This session comprised three national presentations and a summary by a rapporteur of seven posters relevant to the subject of storage containers.

The first presentation concerned the licensing of the Castor 440/84M cask for transport and storage of WWER-440 spent fuel in the Czech Republic.

Several important modifications were identified in the design of the new cask and are related to improved neutron shielding properties, modified trunnion construction, optional use of a third welded lid and new fuel basket design.

The licensing procedure lasted about two years and included the preparation of independent reviews on: cask inventory and criticality calculations, thermal calculations, shielding analyses and radiation safety, the confinement system, and mechanical analyses of the cask and its components.

This procedure concluded with the issuance of the cask licence for railroad transport and storage in 2005.

The second presentation was concerned with research and development for fuel transport and storage in France related to the improvement in fuel design associated with the increase of reactor performance. This is notably the case for evolutionary reactors such as the European Pressurized Water Reactor (EPR), but also for existing reactors. Spent fuels are no longer the same, and include higher enrichments and high burnup fuels (over 60 000 MW·d/t HM) with high heat load and high radiation levels. The management of spent MOX fuel is also a challenging matter.

The current research and development programme is focused on the following items:

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(a) New basket designs and materials with high boron content for sub-criticality (metal matrix composites);
(b) High performance thermal transfer systems;
(c) New neutron shielding materials resistant to more severe conditions;
(d) New SACs to keep the g loads low in all conditions.

The third presentation described the results of full scale demonstration tests on two types of cask for spent fuel in Japan. The tests included heat removal tests and structural integrity (drop) tests.

A new research programme to verify metal cask integrity under long term dry storage conditions has been started. It includes a drop test without impact limiters under accident conditions and metal cask containment performance during long term storage.

The oral and poster papers in this session reflect the trends in container technology. They are connected to the improvement of reactor performance (increasing burnup and the use of new fuel) and the safety requirements for spent fuel management, especially long term storage.

New designs and technology are being developed, as indicated in the paper from France, to cope with advanced fuel and improved reactor performance.

Test programmes to demonstrate container performance are still being carried out as shown in the papers from Japan. This programme reflects the new requirements that come from the increasing storage times for spent fuel.

In addition, there is a desire to improve existing casks to cope with improved reactor performance and this may also be considered to be one of current trends in container technology.

SUMMARY OF SESSION 6

FUEL AND CLADDING PROPERTIES AND CLADDING BEHAVIOUR*

Chairperson

W. GOLL
Germany

The session comprised a keynote presentation, three national presentations, a report by the chairman on two posters relevant to the topic of the session and a summary panel session. The keynote presentation was concerned with the logic that is behind the definition of 'damaged fuel' as given in international documents. It was pointed out that 'damage' is not an intrinsic property of the fuel. Instead, classifying a fuel rod as damaged should depend on the associated system, storage and/or transport condition, and requirements. Examples were given of where the classification of the fuel depends on the atmosphere or mechanical constitution during storage or transport.

A presentation from Ukraine showed the approaches that are used there to classify damaged spent nuclear fuel (DSNF) for long term storage. For interim storage, the damaged fuel is visually characterized and chemically analyzed for I-131, Cs-137 and Np-239 to assess the condition of the cladding. In the next step, the DSNF is retrieved, packed and dried. The presentation included the selection criteria for appropriate storage containers and the measures to ensure operational safety.

A presentation from Germany showed that dry storage plays an important role in spent fuel management, and that its role will increase over the next several years. Already, 100 casks are stored in the interim storage sites in Germany. At the same time, the discharge burnup of fuel is increasing. To take this development into account, creep experiments have been started to assess cladding strain and failure behaviour under the new demanding stress and

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temperature conditions. The tests are geared towards elucidating the mechanism of cracking under long term stress conditions.

Tests on hydride reorientation and on the mechanical properties of irradiated BWR and PWR cladding materials after reorientation were reported in a presentation from Japan. The hydride effect evaluation test matrix covers stresses up to 130 MPa and temperatures up to 400°C with cooling rates of 0.6–30°C/h. The mechanical tests consisted of ring compression and longitudinal tensile tests performed after hydride reorientation testing.

A report was given by the Chairman on two posters from Iraq and Lithuania. Both dealt with the characterization of spent fuel. The Iraqi paper presented a fast evaluation method to overcome the influence of low and high level impurities on the analysis of U-235/U-238 by thermal ionization mass spectrometry (TIMS). The new method replaces the time consuming purification process; it provides improved accuracy and reduces the time required for sample preparation, regardless of the level of impurities present. The Lithuanian paper presented an estimation of the radionuclide content of RBMK-1500 spent fuel using the SCALE 5 computer code. The results were compared with available experimental and numerical data for irradiated RBMK fuel. Numerical and experimental data were in good agreement and demonstrated that SCALE 5 is quite promising for the determination of the RBMK fuel characteristics.

In Summary Panel 2 for sessions IV, V and VI the following points were raised by the panel members and the audience:

(a) Intact fuel:
- Dry storage is a well proven technology;
- The cladding integrity assessment methodology is well established up to high burnups.
- For very stringent cladding conditions with regard to stress, temperature or hydrogen content, the experimental database is still under development;
- The experimental database for storage and transport should be reviewed with regard to its relevance to the short term or the long term, and fuel type (UO_2, MOX);

(b) Damaged fuel — further consideration has to be given to:
- Single fuel rods that lose gas during cask loading;
- Classification of damaged fuel in terms of the functions it has to fulfil in a given phase of the nuclear fuel cycle.

DISCUSSION

C. GANGULY (IAEA): If you have to store the fuel for, say, 50 to 100 years, you should consider the influence of corrosion, because for wall thicknesses of 0.3 to 0.4 mm that could be one of the life limiting factors. Has any recommendation been made to look into thin wall cladding corrosion behaviour for long term storage?

W. GOLL (Germany — Chairperson of Session 6): That point was not addressed, but of course it belongs to the possible aspects that have to be considered.

LOOKING TO THE FUTURE*

Chairperson

A. MARVY
France

I would like to begin by considering some of the factors on a global scale that may influence our work in spent fuel management. One particular matter that has been discussed at this conference is the possibility of a rapidly growing demand for nuclear energy that might happen quite soon. Another matter is the concern for sustainable development that is completely modifying industrial strategy and practice in many countries around the world. This concern is being accentuated by the evidence that there may be some coming shortages of natural resources and so the natural resource that we are concerned with, uranium, is something that we should try to conserve in the coming years. Also, the Chernobyl syndrome is still alive. Its 20th anniversary was not long ago and this is still creating public fear and public reactions. This syndrome could go on — probably depending upon how we behave as an industry and as a community. Also, and this was mentioned early in the conference, a number of dramatic changes have come about since the 9/11 event in the USA, the concern over weapons of mass destruction and also over terrorism. Another matter is public involvement. There is more and more demand for public involvement. I think in the past we were lacking transparency about our activities and this is probably still the case, although the situation has improved quite dramatically over time. It is a topic that we should keep in mind and we should try our best to address it in the coming years.

Now, I would like to shift your attention to the numbers — those numbers with which we started our discussions at this conference. Firstly, we have heard about the plans to build new reactors in the coming years. We have heard that somewhere between 1000 and 1500 reactors could be built by 2050. That is a

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very large number compared to the numbers we have today. Whether this is realistic and whether it can be achieved, I do not know, but I think we should keep those numbers in mind. They have an important bearing on the efforts we need to undertake in order to meet the challenges ahead.

The second set of numbers impresses me the most. They are the predictions of the quantities of spent fuel generated. I think the official number from the IAEA is a forecast of 450 000 tonnes by 2020. That time is not far ahead; this is 2006, and so it is not a very long time. The existing quantities today are close to 200 000 tonnes, so it means that a doubling of the existing amounts could take place within 15 years! That is a dramatic increase and I put between brackets and with a question mark one figure that was also mentioned during the conference: the 1 million tonne mark by 2050. One million tonnes of spent fuel, can you imagine this? It is a large amount and I think Mr González was trying to figure out how many becquerels it would translate into. I think the important questions for the future are probably linked to this number.

Another important number is two. We have at hand only two ways of management, two real management options for spent fuel, reprocessing or disposal. The IAEA and the OECD/NEA have stressed that storage is not the final solution. So we have only two management options left, reprocessing or disposal. We have been discussing disposal quite a lot within the IAEA and the NEA. We have discussed spent fuel reprocessing and the problems associated with it to a much lesser extent. So maybe it is time to reconsider this and I believe that the international organizations should do that.

Another number is one — one temporary management step that is used everywhere around the globe. It is storage. We have been listening to interesting discussions and comments on storage and problems existing with storage systems and also we have been listening to presentations showing that storage is a good solution for a certain period of time but with a question mark about longer times.

And the last number is zero, that is there is no working repository for spent fuel or high level waste (recognising, however, the WIPP geological repository for long lived low level waste in the USA).

So what can we do? I think we can do a lot and there are things that are happening now. We have heard that international proposals are being put forward such as GNEP and the Russian President's initiatives, the MNA, the GIF, and the IAEA's INPRO. All of these initiatives will contribute to a safer use of nuclear energy. We have heard about the US position, that the drivers for GNEP are twofold: waste and non-proliferation. There are many other aspects but I think those two drivers are of key interest.

On safety, the Joint Convention, standards and guidelines have all been mentioned. I think that in this area we can certainly make sure that the international community considers that several things are possible or that different ways can be shared in order to ensure a safer use of nuclear energy. It will also contribute to clarifying the situation in the eyes of the public. We have been talking about an international regulatory regime and about whether this should be developed alone or in parallel with the new designs of reactors and fuel cycle systems. I firmly believe that these two initiatives should be conducted in parallel so that any progress in one field would be reflected in the other. In this way we will find a common view and the best design for the future.

In the field of reducing waste radiotoxicity there are a number of developments and it is an important topic, but it lies in the future. However, there are things we can do at the present time such as fabricating and using MOX fuel, starting the depletion of plutonium in the spent fuel inventory and possibly also using some uranium — the depleted uranium that is readily available — then we would be giving consideration to conserving natural resources. In the future, and we have heard about that too, new nuclear fuel management schemes could become available with the advent of newly designed reactors that would use fast neutron technology and that could also burn all actinides.

I believe that for all of these to be successful we need to go forward and expand our knowledge in various areas in order to achieve whatever is required. I also think that the studies in the many fields should be conducted through international cooperation. I think that our community has a lot to gain through international cooperation and all of this will then substantiate the safety basis for what is required in disposal systems.

I have not mentioned the reprocessing, the necessary extension of storage life and the new reactors. I noted during the week the information on the accumulation of spent fuel and the discussion we had about the term limit for its storage. Once we have put spent fuel into storage we have no idea as to when we are going to remove or retrieve it and so it leaves things open ended. I think that this raises real questions. So we need to expand the studies on the long term behaviour of spent fuel whether because of the higher burnup of most fuel or because, in the future, fuel could also include actinides. Also, how will new materials used in fuel fabrication behave under irradiation, different kinds of spectra or different conditions? I would like to mention the new nuclear systems, and I mean systems — not only reactors or fuel or storage. We should look at all the components together as a system so that we have the opportunity to optimize all of these in order to achieve a much better goal in the future.

The last things I would like to mention are quite important as well; they have been mentioned a few times. One is to involve the public, share information, open our knowledge to whoever is ready to listen to us and also to care for the future through the training of competent personnel and engineers so that we can conduct activities safely in the future.

DISCUSSION

A.J.M.L. MACHIELS (United States of America): With regard to the two options, I would formulate it a bit differently, not saying ‘disposal or reprocessing’. I would distinguish between the direct disposal of spent fuel, which is one option, the other option being reprocessing and recycling with disposal of the waste. In the USA there have been some discussions on this that give the impression that the reprocessing option does not need a geological repository. In fact, whatever scheme is adopted there will always be a need to dispose of the residual waste in a geological repository. So just for clarity I suggest that you rephrase that a little bit.

A. MARVY (France — Chairperson of Session 8): I agree with you.

A. GONZÁLEZ (Argentina): I would like to make a statement about the follow-up to this conference. I hope that, pursuant to this conference, international actions relating to the technical and safety aspects of spent fuel management will be taken on the basis of what we have discussed during this week.

J. BOUCHARD (France — President): I agree with Mr. González that we need to look beyond this conference to the future of international activities in the field of spent fuel management.

CONCLUSIONS OF THE CONFERENCE PRESIDENT*

J. Bouchard
Commissariat à l'énergie atomique,
Paris, France

The scope of this conference was broadened, as compared to the previous conferences on spent fuel organized by the IAEA, to include in its scope the policy, safety and security aspects of spent fuel management. This is a reflection of the increasing importance of spent fuel management and of the recognition of its fundamental place in the future development of nuclear power.

Now is the right time to discuss the strategic aspects of spent fuel management as there is currently ample evidence of a renaissance of nuclear power. For reasons concerned with ensuring secure national energy supplies, limiting the increase of energy costs and avoiding possible global warming by reducing carbon emissions, there are movements in several countries towards a regeneration of nuclear power.

The third generation of reactors will be based mainly on LWRs, but with improved safety and economics. The same back end fuel cycle issues as with current power plants can therefore be expected. This implies that the already large amount of spent fuel in storage will increase dramatically if no choices are made on spent fuel management strategies.

Spent fuel is still differently regarded by countries — as a resource by some and as a waste by others — and the strategies for its management vary, ranging from reprocessing to direct disposal. However, in both cases a final disposition solution is needed and it is generally agreed that disposal deep in geological formations is the most appropriate solution.

In all countries, the spent fuel or the high level waste from reprocessing are currently being stored, usually above ground, awaiting the development of geological repositories. And while the arrangements for storage have proved to be satisfactory and have been operated without major problems, it is generally agreed that these arrangements are interim, that is they do not represent a final solution.

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Although the current arrangements for storage are working satisfactorily, it is becoming increasingly important to have final disposal arrangements available so as to be able to demonstrate that nuclear power is sustainable and that it does not lead to an unsolved waste problem. The conference was updated on the good progress in several countries towards the development of geological repositories — expected to become available after about 2020.

Recent fuel cycle initiatives by the USA and the Russian Federation point in similar directions and have similar overall goals:

(a) Improving control over the increasing amounts of spent fuel;
(b) Helping to reduce proliferation and security risks;
(c) Assisting new countries to develop nuclear power.

They rely on reprocessing and recycling, but with advanced technologies to reduce the proliferation risks and to minimize the generation of radioactive waste.

In 2003, IAEA Director General M. ElBaradei set up an international expert group to take a fresh look at multilateral approaches. While many larger countries may wish to continue to solve the problems of the nuclear fuel cycle, including waste disposal, themselves, multilateral solutions may make economic sense to smaller countries. The multilateral approaches also promise better assurances of security and proliferation resistance. It was proposed that the international agencies should continue to be involved and to evaluate these approaches further and it was also suggested that the IAEA could be a monitoring agency to oversee the safety and other aspects of any multilateral initiatives that may be implemented.

The concept of an international safety regime has developed over the last decade mainly as a result of the coming into force of the legally binding nuclear related conventions prompted by the Chernobyl Accident in 1986. In particular, the Joint Convention (the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management), together with the International Safety Standards may be seen as providing a framework for safety at the international level in the area of spent fuel management. The Joint Convention is legally binding on its 41 contracting parties and requires that spent fuel and radioactive waste management are conducted with regard to accepted norms of safety. The safety norms are derived from the recommendations of the international safety standards, which establish best safety practices based on worldwide experience in the field.

The conference noted that the Joint Convention is an incentive convention and that, at the time of the Second Review Meeting of the Joint

Convention, the contracting parties were not yet prepared to go in the direction of a more mandatory mechanism.

The transport of radioactive material, including spent fuel, represents a particularly good example of the international safety regime. The regulations for safety in transport in each country and by each international mode of transport are drawn directly from the IAEA Transport Regulations. The safety record in the transport area has been exemplary as evidenced by the entirely positive results of spent fuel transports over several decades. An additional element of the safety regime is an international appraisal mechanism. Several countries have invited IAEA teams to conduct peer reviews of their arrangements for transport of radioactive materials on the basis of the IAEA Transport Regulations and guidance. Within the international safety regime, the transport area is an exception, with all countries following common international regulations. This example demonstrates how countries can move towards an international safety regime when all agree on the policy and on the various safety aspects of the application.

There is an obvious linkage between the proposed multilateral fuel cycle initiatives and the international safety regime, that is, any multilateral fuel cycle activities that may be conducted would be expected to comply with the requirements of the Joint Convention and with the recommendations of the international safety standards.

It was noted that the international safety standards in the area of spent fuel management are in the process of being updated and elaborated to cover a wider scope and, during the conference, proposals were made on topics that warrant the development of new safety standards.

Many technical aspects of spent fuel storage were also reviewed during the conference.

At a time when there is much interest in saving space in storage facilities by denser packing, the discussions on the advantages of burnup credit were very topical. Burnup credit means making use of the change in the isotopic composition of fuel, and hence its reactivity, due to irradiation. The presentations at the conference pointed out the substantial benefits that can be obtained from the application of burnup credit. However, much of the assessment and development work on this subject has been done in relation to PWR and BWR fuels and it was clear from the discussions that there is a need for the work to be extended to WWER and RBMK fuels. It was suggested that the IAEA might be able to help in this area.

Most spent fuel storage systems were designed for short term application pending reprocessing or disposal. The time period requirements for storage systems have been extended in most countries because of the unavailability of disposal facilities. In some countries, new facilities have been built for this purpose; in others, the use of existing facilities is being extended for longer periods. An important safety issue is how to establish the safety of the facilities for long term storage. There must be confidence in the continued integrity of the fuel, its container, the structure of the waste store, the maintenance of subcriticality, etc. How can this be achieved? It must be through a combination of monitoring, inspection and research. There was much discussion on this subject and it is clearly an area where more research and regulatory work has to be done.

There is a trend towards dry storage which was clearly illustrated by the presentations at the conference. While the specialists expressed confidence in the technical development of storage facilities and containers in order to satisfy long term needs, it is clearly necessary for more research and development on fuel behaviour in dry storage. In particular, it was mentioned that high burnup fuels and MOX fuels will need to be carefully assessed in the context of ensuring long term storage safety.

Looking to the future, the presentations at the conference show some clear tendencies, which can provide a basis for more international cooperation:

(1) The need for geological repositories for radioactive waste;
(2) The development of advanced reprocessing;
(3) The burning of actinides in fast reactors;
(4) The necessity to increase the duration of interim storage;
(5) The unavoidable increase of transport of both spent fuel and radioactive waste

There is an expectation that there will be a follow-up to this conference by the international agencies. In my opinion it should be in the form of a greater international cooperation on research and development related to the trends indicated above and it should also be in continuing progress towards an international safety regime or at least harmonized safety regulations. It should also be by promoting and monitoring future multilateral initiatives related to fuel cycle activities.

CLOSING ADDRESS

T. Taniguchi
Deputy Director General,
Department of Nuclear Safety and Security,
International Atomic Energy Agency,
Vienna

On behalf of the IAEA I would like to congratulate you on a successful conference. Nearly 200 participants from 41 countries and from organizations with varied nuclear experience have provided us with useful lessons and strategic ideas.

First of all I would stress that the conference marks a major shift for international cooperation in the management of spent fuel from nuclear power reactors. Three years ago, as reflected in the IAEA's conference of that year, spent fuel management was seen as largely a technical matter related mainly to transport and storage technology. However, as underlined by the title of the first session on Monday, the evolving international scene has made spent fuel management one of the important focuses in the growing high expectation for the future generation of nuclear energy in the world.

To support countries in meeting future energy demands and to develop a peaceful, safe and secure nuclear industry worldwide, important international initiatives, clear national strategies and relevant examples are needed, and evidence of them was provided at this conference. A comprehensive and flexible policy approach is needed for spent fuel management and the rigid dichotomy of the past between resource and waste or storage and disposal is to be avoided.

Important international initiatives related to the back end of the nuclear fuel cycle and to the prevention of nuclear proliferation, such as those of the USA and of the Russian Federation, have emerged and were discussed at the conference. However, as stressed in particular by B. Pellaud, any international initiative, any multinational approach needs to be time tested and to have proven technology, as well as a harmonized safety context, before it can be developed. He clearly suggested that the IAEA should have a role if there is to be any development in this context. The challenge in this context is in substantiating these initiatives in an effective, constructive, practical and equitable manner for the common interest of all.

The bases for an international nuclear safety regime were discussed during the conference. In particular, the nature and the membership of the

Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management were discussed. Is the incentive nature of the Joint Convention sufficient to maintain a high level of safety in spent fuel management? Do we need a more effective international framework with firm requirements and reassurances of compliance? Do we need more rigour in the reporting process? According to the President of the second review meeting, the Contracting Parties to the Joint Convention are not yet ready to go in that direction. However, listening to your debate, I conclude that it should be reported to the Contracting Parties for consideration.

Increasing the number of States party to the Joint Convention is an issue that needs attention. One of the conclusions of the second review meeting of the Joint Convention was that an increased membership was needed and it has also emerged from your conference. The IAEA has already prepared an action plan to explain and to convince its Member States of the advantages of ratifying the Joint Convention.

The IAEA Safety Standards complement the international safety conventions and are an important part of the global nuclear safety regime. They are developed and applied in different ways by and for the IAEA Member States and they are recognized by several other international organizations. They serve as a solid basis for the self-assessment by States of their programmes and for international appraisals of safety. The IAEA Safety Standards series of publications is now mature, with the finalization of the unified fundamental principles. However, it is clear from the conference that the series needs to be completed by more guidance on spent fuel management, for example on storage and on criticality safety.

The worldwide application of the IAEA Transport Regulations was shown to be a good example of an effective framework that provides for a harmonized international regulatory infrastructure. The conference highlighted the trends in spent fuel management that can influence both storage and transport safety, such as higher discharge fuel burnup and longer term storage.

Although discussion was limited on this aspect, security and protection against terrorism has become another important area since 11 September 2001, in all areas of spent fuel management including transport.

From the discussions of the conference on the subject of spent fuel storage it is clear that, while there is confidence in the reliability of currently operating storage systems based on wide and long historical experience, there is less certainty about the storage of spent fuel for extended periods into the future. This is especially true of stored MOX and alloy fuels, and fuels with high burnup, for which less data exist on time dependent behaviour. This is an area requiring more research and development in order to provide the necessary confidence to allow such stores to be licensed for decades and up to one hundred years. The IAEA has work in hand on this subject aimed at helping Member States examine and resolve the problem, and I think that the results of this conference will give an impetus for enhanced efforts in the area. As concluded by the conference, the IAEA's Safety Standards need to be strengthened to support national regulatory bodies in licensing spent fuel stores for extended periods into the future.

The subject of burnup credit has been addressed in several IAEA meetings and publications in the past. It is an important approach by which some of the problems of space shortage in reactor pools and in other stores can be alleviated. The discussions at the conference have indicated that, while good information exists on PWR and BWR fuels to allow allowance to be taken of burnup credit, the same information is deficient for other types of reactor fuel, for example WWER and RBMK fuels. It was suggested that the IAEA might be able to facilitate the transfer of this information if it exists, or to encourage research to obtain it. In the same way as for long term storage, there is also a regulatory side to this subject and consideration should be given to the licensing of storage systems that use burnup credit and to review its safety standards in this context to ensure that appropriate guidance is given to regulators.

The work of the conference also highlighted the need for reprocessing and recycling plutonium and minor actinides in fast reactors for efficient utilization of uranium raw material and for minimizing decay heat, waste volume and radiotoxicity of waste for final disposal. In this area, there is a need for international cooperation to improve the technology and to make it efficient, safe, secure and proliferation resistant. The INPRO Joint Study on fast reactors and closed fuel cycle is a typical example of such cooperation.

Last but not least, I would like to express my sincere thanks to all of the people who supported the successful planning and implementation of this conference. In particular, the excellent leadership by the President and session chairpersons, and the coordinated preparation by the programme committee are undoubtedly behind the success of the conference and they deserve great appreciation.

Finally, I hope that you have found the presentations, discussions and the exchange of information of this week useful and that it has contributed in some way towards helping you in your work on this subject. I wish you a safe return to your home countries. I now declare this conference closed.

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Session 1.B	A. GONZÁLEZ	Argentina
Session 2	J. WHANG	Republic of Korea
Session 3	W. BRACH	United States of America
Session 4	T. SAEGUSA	Japan
Session 5	N. TIKHONOV	Russian Federation
Session 6	W. GOLL	Germany
Session 7	A. MARVY	France
Session 8	J. BOUCHARD	France

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National strategies for the management of spent fuel vary, ranging from reprocessing to direct disposal. This is indicative of the fact that spent fuel is differently regarded by countries — as a resource by some and as a waste by others. At the moment most spent fuel is in storage at nuclear power plant sites, at a few centralized storage sites and at reprocessing facilities. The next steps towards the disposition of spent fuel are either reuse, through reprocessing, or disposal in geological repositories. Because progress on implementing these strategies is slow in most countries the amounts of spent fuel in storage are increasing. The management of spent fuel is, for strategic, economic, safety and security reasons, a key issue for the future of nuclear power and is an issue that many States have yet to decide upon. The International Atomic Energy Agency organized an international conference on Management of Spent Fuel from Nuclear Power Reactors to provide an opportunity for the exchange of information on the subject among Member States and to look for common approaches to the issues identified. This publication, which constitutes the record of the conference, includes the opening and closing speeches, the keynote papers, the summaries of the panel discussions and sessions, the Conference President's summary and an executive summary. A CD-ROM containing the unedited contributed papers to the conference can be found at the back.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 92–0–102107–0
ISSN 0074–1884