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UNEDITED PROCEEDINGS

International Conference
held in Vienna, 23–26 June 2003
organized by the International Atomic Energy Agency
in co-operation with the Electric Utility Cost Group Inc.,
International Science and Technology Centre,
World Energy Council and
World Nuclear Association

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FOREWORD

Nuclear power is a significant contributor to the global supply of electricity, and continues to be the major source that can provide electricity on a large scale with a comparatively minimal impact on the environment. But it is evident that, despite decades of experience with this technology, nuclear power today remains mainly in a holding position, with its future somewhat uncertain primarily due to concerns related to waste, safety and security. One of the most important factors that would influence future nuclear growth is the innovation in reactor and fuel cycle technologies to successfully maximize the benefits of nuclear power while minimizing the associated concerns.

The International Conference on Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power, organized in June 2003 by the IAEA in co-operation with the World Nuclear Association, the World Energy Council, the International Science and Technology Center and the Electric Utilities Cost Group, comes at a pivotal point in the history of nuclear energy. Progress has been made recently on many fronts related to nuclear power, including waste disposal, license extension, and safety and security upgrades. Many countries are engaged in innovation projects and while most of the current expansion in nuclear energy is taking place in East and South Asia, recent years have witnessed statements and actions in North America, Europe and elsewhere that support a renewed consideration of the merits of nuclear power.

The conference included talks on specific topics by 21 invited speakers drawn from 11 Member States as well as 21 oral presentations and 26 poster presentations of accepted papers. Part of the opening session and two half-day sessions were devoted to panel discussions in which 23 panelists from 9 Member States and 5 international organizations took part. All relevant aspects of innovative technologies for nuclear fuel cycles and nuclear power were discussed in an open, frank and objective manner. These proceedings contain a summary of the results of the conference, invited and contributed papers, and summaries of panel discussions.

The IAEA wishes to thank the invited speakers, panelists, authors, programme committee members and all the participants for their contributions in making this conference a success. The IAEA officers responsible for this publication were V.M.R. Koorapaty and J. Kupitz of the Division of Nuclear Power and F. Fukuda of the Division of Nuclear Fuel Cycle and Waste Technology.

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SUMMARY

1. INTRODUCTION

In response to the Agency's General Conference Resolutions GC(44)/RES/21 and GC(45)/RES/12, both entitled "Strengthening the Agency's activities related to nuclear science, technologies and applications" drawing attention to the statutory requirement for the Agency to "encourage and assist research on, and development and practical application of, atomic energy for peaceful uses and foster the exchange of scientific and technical information", the Agency organized the International Conference on Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power, in Vienna from 23 to 26 June 2003, in co-operation with the World Nuclear Association, the World Energy Council, the International Science and Technology Center and the Electric Utilities Cost Group.

2. CONFERENCE OBJECTIVES

The main objectives of the Conference were to facilitate exchange of information between senior experts and policy makers from Member States and international organizations on important aspects of the development of innovative technologies for future generations of nuclear power reactors and fuel cycles; to create an understanding of the social, environmental and economic conditions that would facilitate innovative and sustainable nuclear technologies; and to identify opportunities for collaborative work between Member States and international organizations and programmes.

3. CONFERENCE STRUCTURE

The total duration of the conference was three and one half days and was structured to maximize the time for discussion of important issues. There were seven sessions of different durations with no parallel sessions. Four sessions were devoted to talks on specific topics by 21 invited speakers from 11 Member States. One full-day session included to 21 oral presentations and 26 poster presentations. Summaries of the posters related to nuclear power and nuclear fuel cycles were also included in the oral presentations. Part of the opening session and two half-day sessions were devoted to panel discussions in which 23 panelists from 9 Member States and 5 international organizations took part. A total of 158 participants from 35 countries and 9 international organizations attended the conference. Seventy of the participants were from 22 developing countries, sixty seven from 13 developed countries and twenty one from 9 international organizations.

All relevant aspects of innovative technologies for nuclear fuel cycles and nuclear power were discussed in an open, frank and objective manner, and a summary of the observations and conclusions are presented in the following paragraphs.

4. OPENING ADDRESSES AND PANEL ON CONFERENCE THEME (SESSION 1)

The opening speech of the Director General, IAEA, read out by the Deputy Director General, V. M. Mourogov, touched upon the contradictory conditions in which nuclear power finds itself. Nuclear power is a significant contributor to the global supply of electricity, and continues to be the major source that can provide electricity on a large scale with a comparatively minimal impact on the environment. At the same time, it is evident that despite decades of experience with this technology, nuclear power today remains mainly in a holding position, with its future somewhat uncertain primarily due to concerns related to waste, safety and security. One of the most important factors that would influence future nuclear growth is the innovation in reactor and fuel cycle technologies to successfully maximize the benefits of nuclear power while minimizing the associated concerns. In the words of the Director General *"This conference comes at a pivotal point in the history of nuclear energy. For nuclear technology to make a substantial contribution to energy supplies, innovation is essential — innovation that is global in scope, responsive to concerns, and collaborative in its approach".*

The main points arising out of the panel discussion on the conference theme Chaired by L. Echávarri are the following. The demand for electricity and other sources of primary energy may result in a revival of interest in nuclear energy provided it fulfils the requirements of a competitive market scenario. The global nuclear community is willing to coordinate efforts for the development of innovative technologies. It is a sign of maturity that developing and developed countries are discussing this matter together. The need for education and preservation of nuclear experience and knowledge is proposed to be met by the World Nuclear University (WNU) with active support from the IAEA and sponsorship of many private organizations in the world. While the current nuclear power plants are competitive, new plants will face a different scenario and market conditions. In this regard, the EUCG would be willing to assist in meeting the needs for bench marking new projects in terms of cost of development, construction, operation and decommissioning. The ISTC would be willing to collaborate on studies on materials, radiochemical aspects, etc. While the economic aspects of nuclear energy are important, one could not overlook the potential dangers that the rest of the century might face including an energy crisis, global warming and terrorism. The role of the governments will be crucial in developing energy policies as well as measures to avoid the spread of nuclear technology for non–peaceful uses. International Organizations would have a key role to in facilitating the cooperative work.

The perspective of a developing country with an expanding nuclear power program presented by A. Kakodkar, India, explained the rationale for considering utilizaion of thorium in a closed fuel cycle to ensure long-term energy security for the country. A research reactor KAMINI is operating in Kalpakkam based on Uranium-233 fuel, derived from thorium. This fuel was bred, reprocessed and fabricated indigenously. An Advanced Heavy Water Reactor (AHWR) is being developed at the Bhabha Atomic Research Center (BARC) to expedite transition to thorium based systems. India tested the mixed uranium-plutonium carbide fuel in their Fast Breeder Test Reactor at Kalpakkam and achieved all its technological objectives. The detailed design and technology development of the 500 MWe Prototype Fast Breeder Reactor (PFBR) has been completed and construction would start soon. The rapid growth of electricity demand in developing countries may result in demand for nuclear generating capacity in countries that may not be 'nuclear capable'. This should be a matter of global interest in view of its potential to protect the earth from irreversible climate changes. Nuclear energy is based more on knowledge, less on materials, than most other energy sources, and therefore, requires expertise in several disciplines of science and technology. This expertise has to be acquired through painstaking efforts. This spread of nuclear technology can occur only if there is a strong human resource development component, which is a prerequisite for technology assimilation. In this context, the role of the IAEA cannot be overemphasized.

The invited talk on "Innovative Nuclear Fuel Cycle for the Future" presented by Y. Fujiie, Japan, stressed the following points. Global economic development in the 20th century was part was based mainly on large-scale consumption of fossil fuels. This development disregarded the environmental impact from CO_2 generation. The growth of nuclear power industry to a mature stage was based on energy security, the very small amount of waste per unit of electricity generated and the success in the front end of the Nuclear Fuel Cycle. However economic development in the 21st century has to take into account mainly the environment and the CO_2 production under the Kyoto Protocol. The challenge for nuclear science & technology will be to demonstrate the waste management technology, continued safety, prevention of proliferation and cost effective fuel recycling. Assuming that the environmental protection is our responsibility, Japan will continue to expand the use of nuclear energy with MOX fuel and to close the fuel cycle. Also, Japan will explore advanced Nuclear Fuel Cycles and the use of fast reactors. R & D should play a key role in the development of suitable technology and international collaboration should be explored. Japan offers to the international nuclear community and the R & D centers the use of the Monju reactor for shared R & D programmes.

5. NEEDS, PERSPECTIVE AND CHALLENGE FOR INNOVATION (SESSION 2)

L. Echávarri, OECD/NEA, pointed out that this century will have considerable needs for energy, heat, hydrogen, and water. Nuclear energy could play a major role in meeting all these needs. Cross cutting issues regarding R & D needs will present opportunities to share facilities and expertise for these purposes. Several disciplines have to be brought together to meet the requirements of innovative systems on a collaborative bases facilitated by international organizations.

J. Murray, WEC, in her keynote paper on "Energy Demands and the Nuclear Role" referred to some of the setbacks experience by nuclear energy after the release of the Bush Administration's energy policy for the US. The proliferation concern was heightened again due to recent situation in Iran and the Democratic Peoples Republic of Korea. Additionally nuclear energy growth has to occur in a deregulated market. Innovative solutions are needed in all areas. The future world energy growth will not be a straight extrapolation of the present, GDP growth will not be linear, population growth predictions may not come true and technological growth will not be as easy as in the past. Innovation should provide a diversity of offered products; reduced cost without compromise of safety, enhanced security, effective waste management, and enhanced proliferation resistance among others.

H. H. Rogner, IAEA, presented the keynote paper on "Nuclear Power in the 21st Century". One of the paradoxical uncertainties in 21st century energy supplies is the role of nuclear power as its near-term role appears less certain that its longer-term role. With respect to the application of cost targets, the paper emphasized that costs related to safety, environmental impacts, financial risk and proliferation resistance should be internalized in the comparison of nuclear energy to alternatives.

W.C. Turkenburg from the Netherlands presented a keynote paper on "Environmental and Sustainability Aspects". In his view, the current nuclear power programmes are not sustainable due to reasons including the following: inadequate level of public acceptance; lack of trust with regard to the assessments of nuclear experts; safety related incidents mainly in fuel cycle facilities; ultimate disposal of nuclear waste (HLW) could be solved technologically but not emotionally; technical features alone cannot provide a fully proliferation resistant nuclear system; accumulation of radio nuclides such as Kr-85 can influence the electric properties of the environment and the formation of clouds; scarcity of Uranium resources at low cost; and the required safety features impose severe limits on the competitiveness of nuclear power. The challenge for innovation is to reverse these factors.

C. Cantrell, EUCG, presented the trends in U.S. nuclear power production costs. In the future, the U.S. nuclear power industry will face continued upward cost pressures and nuclear plant operators will have to manage safety margins as well as cut costs. Future economic survival will depend on the industry's ability to create innovative solutions for controlling costs through process improvements and tapping increased generation potential.

V. M. Mourogov, IAEA, made the following points in his keynote paper on "Innovation in Nuclear Technology and the Role of the IAEA". Thirteen countries generate more than a third of their electricity with nuclear energy. The role of IAEA is foreseen to be in facilitating the identification of R&D requirements for innovative technologies. This is done mainly through facilitation of exchange of information through meetings; preparation of technical reports; peer reviews; coordinated research projects; training courses and workshops; and establishment and maintenance of information systems. A real challenge will arise if the developing world changes rapidly the energy requirements under new paradigms and innovations would be the answer to bridge the technological gap.

6. EVOLUTION OF SOCIAL, ECONOMIC AND POLITICAL CONDITIONS (SESSION 3)

The keynote paper on “What do societies expect from innovative nuclear technologies?” was presented by B. Barré, France. A majority of people believe that nuclear power contributes significantly to global warming and climate change. Many are convinced that radioactivity has insidious and mysterious ways to percolate through matter and contaminate the environment. When people are told that Iodine 129 has a "life" of 16 million years, they are scared instead of thinking how little radioactivity this element emits. It is very hard to convey the notion that the health effects of radiations depend only from dose and dose rate, irrespective of whether it is "natural" or "artificial" radiation. There are at least two universal expectations from innovative reactors. The first demand is the guarantee of no significant release of radioactivity in the environment under any circumstances. The second expectation concerns almost unanimously the radioactive wastes: less waste, less long-lived wastes, and no waste at all if possible. However the most significant expectations are expressed in the following wishes of common people: “I expect power when I flip my switch” and "Do not increase my electricity bill”.

The paper on “Technical Aspects of Innovative Nuclear Systems Including Reliability and Safety” was presented by F. Eltawila, USA. Several evolutionary concepts have been presented for the licensing process. These concepts respond to the present technology with new safety and layout changes that allow NPP to address near term requirements. Advance Reactor designs incorporate Safety Research and operating costs improvements. The IRIS project is been supported mainly because of its Passive System reliability. The innovation effort should be concentrated on the improvement of the public impression. Safety, waste, etc should have “zero risk”. That means no accident, no waste. It is important to keep the public informed and educated.

The same topic was covered from the Russian perspective by A.M. Dmitriev. International collaboration between licensing authorities (licensing infrastructure) and the development of mutually accepted underlying principles are required for large scale deployment of nuclear power. Reactor using a solid coolant was considered as an innovative option and was described in detail.

I. S. Chang, Rep. of Korea presented the paper on “Environmental Effects Including All the Stages of the Fuel Cycle”. Most of the innovative fuel cycle concepts explicitly focus on the back-end and aim especially at dealing with the remaining waste. The main characteristics of the innovative fuel cycle concepts are to introduce additional waste management options, such as partitioning and transmutation, in order to reduce the mass and radioactivity of wastes going for final disposal. They are trying to close the fuel cycle not only for plutonium but also for the minor actinides. Compared with the results of the conventional fuel cycle options, on the whole, the innovative fuel cycles have much more benefits in terms of natural uranium use and spent fuel amounts. It indicates that the LWR + FR(Homo), LWR + FR(Hetero), 100% FR and Double Strata options have 34%, 55%, 100%, 42% uranium resource savings, respectively, compared with the once-through option. The 100% FR option requires only depleted uranium for making nuclear fuels instead of natural uranium. In the case of the spent fuel amounts to be disposed of, it is found that spent fuels from the LWR + FR(Homo), LWR + FR(Hetero), 100% FR and Double Strata options could be reduced by 39%, 47%, 71% and 99%, respectively, compared with the once-through option. It indicates that the 100% FR option requires the smallest natural uranium resources while the Double Strata option generates the smallest spent fuel amounts.

K. Samejima, Japan presented the paper on "Economic Viability of Innovative Nuclear Reactor and Fuel Cycle Technologies". The relative competitiveness of nuclear power, which varies depending on the regions and could be enhanced significantly by climate change regulations, is critically important to assess future potential growth of nuclear power. For a liberalized electricity market, average life-long power generation cost may not be the best criteria for utility investments. Short-term return on investment (such as return on asset and cash flow) may become critically important. As a result, small and modular-type reactors could become the preferred designs for low growth and/or relatively small grid markets. But large reactor designs can continue to be preferable for high growth and/or large grid market. Therefore it is important to assess market needs more carefully in determining the future advanced reactor designs. International dialogue among various stakeholders is essential to understand better the requirements of future, next generation nuclear reactors.

T. Rauf, IAEA, presented the keynote paper on "Political Factors including Proliferation Resistance and Nuclear Security". The NPT recognizes rights and establishes obligations regarding the reach, production and use of nuclear energy. Art IV of the NPT and Agency's functions under the technology pillar require trust in order to function optimally. The Agency's fundamental role is to build the kind of trust that facilitates peaceful co-operation in the nuclear field. The major challenge for the non proliferation regime today remain to be the lack of complete trust with respect to States intentions regarding their nuclear programmes, and continuing advances in nuclear technologies for enrichment and reprocessing.

The development of truly proliferation-resistant nuclear energy systems could potentially make a major contribution to overcoming lack of trust. However it was agreed that the degree of proliferation resistance results from a combination of, *inter-alia,* technical design features, operation modalities, institutional arrangements and safeguards measures. The combination of *intrinsic proliferation resistance features* and *extrinsic proliferation resistance measures* are foreseen as a big improvement for the regime and will guarantee that *lack of trust does not result in technology-denial*.

7. PANEL ON "CHALLENGE FOR THE DEPLOYMENT OF INNOVATIVE TECHNOLOGIES" (SESSION 4)

R. O. Cirimello and J. B. Ritch chaired the panel discussion. In his introductory remarks, Cirimello pointed out the following characteristics of innovative designs as challenges arising from the previous three Sessions:

— Inherently safe (no accidents, no explosions)
— With a containment that can withstand internal or external incidents (plane crash, earthquake, terrorist attack, retain all radioactivity inside in any case)
— Built underground (visual impact, less shielding)
— Non-retrievable fuel (resistant to diversion)
— Non-reprocessable fuel or without weapons grade material after its final irradiation period (for non-proliferation)
— Spent fuel should be inherently ready for final disposition (inherent HLW "containment")

— It should compete in efficiency and total cost with CCGT in countries with large gas reserves (electricity cost)
— Demonstration plants should be built to create public trust to promote public acceptance)
— Operation system should be supported by artificial intelligence (avoid human errors)
— Very short maintenance periods and short outages if needed (low operational cost and low personal doses)
— Factory construction (short construction period and low/no on-site specialized work)
— Capable of being decommissioned without dismantling (avoid dispersion of irradiated material)
— Remote technology for radiological and physical protection, safeguards and environmental surveillance. (reliance on advanced technologies)
— Modular and standardized installations (low construction costs)
— New materials and compounds for structural, core and fuel (resistant to irradiation damage, less expensive, high resistance)

A .C. de Oliveira Barroso, Brazil, made the following points: If positive signs of economic development become steady in the next couple of years, then both Angra III (Brazil) and Atucha 2 (Argentina) could be on line by 2012. Then new generation plants could be considered. For Brazil and Argentina, given their technology base, the next generation of water-cooled reactors is more probable. NPPs with sizes adequate to the market are essential (200 – 400 MWe). Countries pursuing nuclear power could be categorized into 3 groups: (a) those with NPPs in operation and under construction/planning (Rep. of Korea, Japan, China, India, etc.); (b) those with operating NPPs but none in construction or ordered; and (c) those with no NPPs but needing new generation capacity in the near future.

Group (a) will probably decide the type of the fourth generation plants. For groups (b) and (c), any decision from the US concerning a new order will be a kind of path breaker, with important influence on group (a) too. Competitiveness will be a decisive factor because: current generation reactors are demonstrating excellent safety and reliability characteristics and almost all the next generation designs promise to go much further in this respect. Also with respect to sustainability, proliferation resistance and physical protection, good advances are being made and these concerns may not rule out the nuclear option in favor of other alternatives. Targets considered favourable are: Capital investment < US$1000/kWe; Production costs < US$30/MWh; Construction period < 3 years.

If international cooperation achieves a level of understanding such that only a few *internationally accepted* recycling or refabrication facilities and final repositories exist and are operated based on shared costs and due royalties (to host Country), then fuel supply and spent fuel return can become a profitable service. Next generation NPPs can be deployed even in countries that do not want to have a “nuclear infrastructure”; This would also take care of most of proliferation and final waste concerns; But the achievement of such a level of understanding and international cooperation may be the ultimate challenge.

The following points summarize the view of R. Duffey, Canada.

Challenges become Opportunities. Meeting Changing Market Conditions / Restructuring / Deregulation / Competitive Power Pools have a strong effect

Old paradigm: Government owned utility and/or oligopoly**,** returns at low discount rates (lower cost of capital), obligation to supply to meet demand, long term view and guaranteed customers.

New paradigm: Large investor/privately owned generating company, higher discount rates to attract investment (higher cost of capital), power contracts with market operator based on price, Short term view with low long-term risk.

Meeting the Long-Term Sustainability Challenge: the future Innovative Plant has multiple product streams. Thorium fuel input, electric power, Hydrogen and process heat plus heavy water, drinking water.

Technology is a race: the challenge is to win. In energy markets, costs dominate for new technology introduction (Photo voltaic, gas turbines, reactors). Technology improves and costs are reduced as markets are penetrated. The trend follows a learning/experience curve (Power Law or Minimum Cost Equation) based on classic economic forces. The curve followed is governed by development costs and market targets. CANDU follows such a curve to compete with other technologies and to be part of international (Gen IV) initiatives. Funding impacts directly the market penetration and on the "learning rate". The CANDU/AECL development path (experience curve) is a chosen balance between evolution and innovation for a competitive advantage.

P. Bernard, France, presented the following scenario of nuclear power:

Needs: Increase of energy demand, Diversification of energy vectors (Electricity, Hydrogen, Cogeneration, Desalination, etc.)

Objectives / constraints: a sustainable development with significant advances in economics, sustainability: minimization of HLW & saving of natural resources, safety and reliability, proliferation resistance & physical protection

These objectives are materialized by the goals of 4th Generation systems deployable by 2030.

Goals & challenges could be summarized as follows:

Economics: Innovative technologies & Innovative options.

Sustainability: minimization of waste; saving of resources; closed fuel cycle & fast neutron spectra.

Safety and reliability: Innovative safety approach coherent with the current practices. Proliferation resistance & Physical protection: Optimization of inherent and extrinsic features

Technical & Scientific Challenges: Innovative technologies & Innovative options; high temperature materials; hydrogen production ; innovative systems for the safety functions ; etc.

Fast Neutron spectra & closed cycles: Innovative fuels and materials; integral actinide extraction and recycling.

Innovative safety approach coherent with the current practices: Gradual integration of the risk informed approach; core melt exclusion strategy; dynamic confinement; etc.

Optimization of inherent and extrinsic features: Integrated reprocessing and refabrication; etc.

J. Vergara, Chile, pointed out that innovative concepts will face the following framework:

Changing rules by vendors may mean a very rapid pre-birth "phase out". Nuclear investment is already "too" risky to take on extra-risks. Unknown prospects for H2 production, water desalinization, etc.

R. K. Sinha, India, expressed that there are four challenges for nuclear energy in the near and medium term:

(1) Matching future technologies with future needs

(2) Developing and qualifying new technologies and design approaches: New materials, associated design approaches and criteria, post irradiation data; nuclear data in uncharted nuclide-energy domains; innovative heat removal systems; validation of codes; acceptance criteria; reliability of passive systems; international collaborative R&D could help cut down costs in all these areas.

(3) The challenges of deploying innovative nuclear energy systems: Managing the transition from the current to the innovative; co-existence of contemporary, evolutionary and innovative designs in the same time frame; licensing of new technologies (The Agency should provide international consensus safety guidelines to assist the MS on the licensability of innovative nuclear fuel cycles and power plant projects, as well as to encourage the MS to streamline the existing licensing process).

(4) Some specific challenges in deploying nuclear power in developing countries:

Most of the programmes are addressing the issues within the framework of a national interest. Even if international in intent, most programmes appear to have goals which are concentrated on meeting the needs of developed countries rather than developing countries.

Y. Sokolov, Russian Federation, explained the country's policy on innovative technology for nuclear energy as follows:

(1) State policy: Closing of the nuclear fuel cycle is a strategic line of nuclear power development in Russia.

(2) Regulations, standards adjustment for innovations.

(3) Inertia of the current NP technologies.

(4) Innovations in mentality; Environmental factors such as radiological vs. industrial risks to human health; deterministic exclusion of severe accidents vs. probabilistic analysis; contradiction of reinforcing the defense-in-depth and economic competitiveness simultaneously; principle of radiation equivalency.

(5) International cooperation: Russia proposed the innovative nuclear system based on fast sodium-cooled reactor with corresponding fuel cycle for testing of the INPRO Methodology during case studies. Assessment of the methodology can be performed jointly by Russian experts and experts from INPRO member-countries. Taking into account that the INPRO ultimate goal is directed to identification of technologies for implementation as an international project and GIF aims at joint international technology development, harmonization of INPRO and Generation IV Programs is needed.

D. Nicholls, South Africa, expressed that the following points ruled the future actions in promoting nuclear energy:

Public Opinion: Safety (demonstrate by actual reduction of emergency planning requirements); waste (demonstrate a solution by example, Yucca Mountain, Finland, etc.); decommissioning (demonstrate this as a routine engineering activity by example)

Economics: Build time (Time from significant expenditure to power generation); cost per KWe installed (Within 10-20% of coal fired alternative); size (small size limits risk to utility); predictability (regulation, operating costs, performance).

Proliferation Resistance: Technology features (can material be easily diverted?); physical safeguards (how to count and account for the fuel elements?); programmatic safeguards (why does a country need a national nuclear laboratory to support power reactors?)

Belief in Ourselves: Nuclear power expansion by 2010, not by 2050! E.g., Calder Hall NPP (3 years from Cabinet decision to power generation and 40+ years of commercial operation)

8. INTERNATIONAL PROGRAMS ON INNOVATIVE NUCLEAR SYSTEMS (SESSION 5)

S. Guindon, Canada, presented the paper on “Status of Work under Generation IV International Forum (GIF)”. The GIF is a formal, government sanctioned organization committed to the collaborative pursuit of R&D on Generation IV systems. Currently, it consists of representatives from ten countries (Argentina, Brazil, Canada, France, Japan, Republic of South Africa, Republic of Korea, Switzerland, United Kingdom, and United States). Members of the GIF foresee benefits from collaborative R&D leading to the selection of new “sustainable” nuclear energy systems. The GIF was launched in January 2000 when the Department of Energy’s (DOE) Office of Nuclear Energy, Science and Technology convened a group of senior governmental representatives from the original nine countries to begin discussions on international collaboration in the development of Generation IV nuclear reactors.

M. Hugon, EC, presented a paper on “EURATOM Research on Future Systems in the Fifth and Sixth Framework Programs”. As one of the energy sources of the European Union (EU), nuclear power had an important role in the debate launched by the Green Paper “Towards a European strategy for the security of energy supply”, which was presented in 2000 by the European Commission. The fifth framework programme (FP5) (1998 - 2002) of the European Atomic Energy Community (EURATOM) has two specific programmes on nuclear energy, one for indirect research and training actions managed by the Research Directorate General and the other for direct actions performed by the Joint Research Centre of the European Commission. Activities are been conducted in three centers: Partitioning and Transmutation in Karlsruhe, Petten for irradiation of fuel and Geel, Belgium, for neutron data and material development. HTR development (DG JRC – IE in Petten) with 21 EC Organizations has 8.45 M euros allocated.

D. Hittner, France, presented the paper on “Michelangelo Network (MICANET)”. The R & D and innovation required for the future of Nuclear Energy needs a large international collaboration. MICANET is the tool for defining by consensus the common objectives in R & D for Nuclear Science and Industrial Applications. On the bases of this common understanding several project of R & D are underway. These projects fill the gaps identified and made the necessary arrangement to be linked with other international initiatives like Gen IV.

J. Kupitz, IAEA, presented a paper on “Background and Structure of INPRO”. The objective of INPRO, which comprises two phases, is to support safe, economic and proliferation resistant use of nuclear technology, in a sustainable manner, to meet the global energy needs in the next 50 years and beyond. During Phase I, work is subdivided into two sub phases. Phase IA focuses on determining user requirements in the areas of economics, environment, safety, proliferation resistance, and crosscutting issues and developing methodologies and guidelines for the comparison of different reactor and fuel cycle concepts and approaches. Work packages contribution were explained.

F. Depisch, IAEA, presented the paper on “Results Achieved within INPRO”. INPRO has now finalized its Phase IA. At its last meeting in May 2003 the Steering Committee approved the Final Report, which was made available at this conference. It reviews the basic principles and user requirements for the areas of economics, environment, safety, proliferation resistance and crosscutting issues and provides a description of the methodology.

D. Nichols, South Africa, presented the paper on “PBMR Program”. The PBMR Programme is led by ESKOM (South Africa Utility) that promotes the use of a concept developed in the 60´s in Germany that uses Helium as coolant and graphite balls containing coated particles of uranium fuel. The name arises from the fact that the balls form a bed through which the gas coolant flows. The project is in the phase of site approval and likely will start the construction of a prototype by the end of this year.

A. Gagarinski, Russia, presented the paper on "U.S.-Russia Collaborative Programme". The Russian President's initiative on energy supply for the sustainable development of mankind, put forward in 2000 at the Millennium Summit, and the U.S. national energy policy announced a year ago are based on similar ideas. The two Presidents at their meeting in May 2002, announced their intention to collaborate in research and development of new, more environmentally safe nuclear power technologies. Establishment of joint expert groups, including the group to prepare recommendations on the joint efforts in innovative nuclear reactor and fuel cycle technology R&D, was the first step in realizing this intention. The group's Russia/U.S. collaboration report identified the prospective directions of such collaboration in the field of innovative reactor and nuclear fuel cycle technologies. An action plan was also prepared, which includes, as the next objective, harmonization of activities within the frameworks of "Generation-IV" International Forum and INPRO.

L. Herrera, USA, presented the paper "U.S. Department of Energy International Nuclear Energy Research Initiative (I-NERI)". In January1997, the President of the United States requested his Committee of Advisors on Science and Technology (PCAST) to review the current national energy research and development (R&D) portfolio, and provide a strategy to ensure that the United States has a program to address the Nation's energy and environmental needs for the next century. In 1999, in response to the PCAST recommendations, DOE established the Nuclear Energy Research Initiative (NERI). Recognizing the importance of a focused programme of international cooperation, PCAST issued a June 1999 report entitled Powerful Partnerships: the Federal Role in International Cooperation on Energy Innovation. It highlights the need for an international component of the NERI programme to promote "bilateral and multilateral research focused on advanced technologies for improving the cost, safety, waste management and proliferation resistance of nuclear fission energy systems. The report further states that the cost of exploring new technological approaches that might deal effectively with the multiple challenges posed by conventional nuclear power are too great for the United States or any other single country to bear, so that a pooling of international resources is needed. In 2001, the U.S. Department of Energy (DOE) established the International Nuclear Energy Research Initiative (I-NERI) to help overcome the principal technical and scientific issues affecting the future use of nuclear energy and to foster global cooperation in nuclear technology.

9. INNOVATIVE NUCLEAR SYSTEMS AND RELATED R&D PROGRAMMES (SESSION 6)

The highlights of the papers and posters presented in this Session are listed below country wise.

Argentina: Reported on an advanced project for an Inherently Safe SPR, an advanced system for Uranium enrichment and an application for desalination.

Armenia: The analysis presented showed that the sustainable energy long-term development in Armenia can be achieved by the utilization of innovative nuclear reactors with a high-level operational safety and economic indicators.

Brazil: Reported a study for fixed bed nuclear reactor (FBNR) using spherical fuel elements constituting a suspended reactor core at its lowest bed porosity. Also its participation in the IRIS (Westinghouse) international project was reported.

Bulgaria: A light water reactor concept is proposed using a very low pressure (4-5 Mpa) in order to minimize the damages caused by of all types of LOCAs.

Canada: Presented the use of the latest design of Advanced CANDU Reactors (ACR) to supply steam for processing oil sands and also to produce electricity. AECL proposed the design as a logical step forward in the CANDU fuel-channel reactor design process to achieve major improvements in economics while expanding safety margins.

Chile: Two advanced ship design proposals are discussed for nuclear propulsion based on data availability**.** Certain large size cargo ships may incorporate reactors such as CAREM-F, IRIS, or MRX. Most high-performance cargo vessels may accommodate reactors similar to the less expensive GT-MHR.

Croatia: An analysis illustrates the applicability of probabilistic method to compare the economics of electrical power generating system. This analysis indicates that the main effort in future innovative reactor design should be more concentrated to improve their operational safety and reliability than to substantially lower their investment costs.

Egypt: Installing a nuclear power plant in Egypt, at the El Dabaa site, to produce electricity and potable water, has the potential to bring major social and economic benefits and will gain the public acceptance and promote utilization of nuclear energy.

France: Project INOVACT in EDF was established to detect nuclear options, which could present an industrial interest for the replacement of the EDF fleet at the two horizons of 2020 and 2070.

India: The three stage Indian nuclear program aims to establish a sustainable nuclear energy system mainly based on closed nuclear fuel cycle and thorium utilization. The second stage aims at setting up of Fast Breeder Reactors (FBRs) for power production and fissile material multiplication.

Indonesia: A Reactor (IEPR) 100 MWe has been designed. The reactor fuel is designed for utilizing Cirene type fuel with coated particles (CFP, Coated Fuel Particles) 4,8% enriched.

Japan: CRIEPI and JNC have started a joint study to implement integrated experiments of electrometallurgical reprocessing of metal and oxide Pu-containing fuel at chemical processing facility (CPF) of JNC-Tokai. An innovative concept for a reduced-moderation spectrum boiling water reactor (BWR) combined with an advanced fuel cycle system (BARS) has been developed as part of the IVNET (Innovative and Viable Nuclear Energy Technology). An innovative water-cooled reactor concept named Reduced-Moderation Water Reactor (RMWR) is under development by JAERI in cooperation with some Japanese utilities and vendors. It is also pursuing the development of the fast reactor (FR) and related fuel cycle system as a priority target to secure its own energy supply. Another paper introduces an activity on Protected Plutonium Production (P3-project) focusing on the potential of light water reactor technology. A study addressed the technical feasibility of reprocessing of HTGR fuels by using the reprocessing plant for LWR fuels. The results of this study showed the technical feasibility of reprocessing of HTGR fuels by using the reprocessing plant for LWR fuels.

Rep. of Korea: the concept of the DUPIC fuel cycle (DUPIC stands for Direct Use of PWR spent fuel in CANDU reactors), is to reuse spent pressurized water reactor fuel as a fuel for CANDU reactors without the reprocessing operations typical of recycling fuel cycles.

Lithuania: In Lithuania, an advanced Reactivity Control Method (RCM) has been proposed for control of the power field of nuclear reactor during its normal operation, start-up and normal and emergency shutdown by the uniform change of distribution of solid neutron-absorbing material concentration in the reactor core using the new spiral Elastic Reactivity Control Device (ERCD).

Netherlands: Comprehensive passive systems that are fully independent of the operator considerably and credibly and decrease the core damage frequency and also the frequency for a large release have been proposed. The use of inherent safety approaches can make the new systems considerably simpler and, thus, safer.

Romania: Competitive levelized cost of generated electricity, larger burn up of discharged fuel and improved safety features are common requirements for all the five reactors of Cernavoda NPP. Lower investment cost, shorter time for commercial operation and advanced safety characteristics are the innovative requirements for the Units 4 and 5.

Russian Federation: A solid coolant, for example, the carbon-based one, is proposed as coolant for the primary circuit of nuclear reactor without excess pressure. Such coolant withstands temperatures up to ~4000K without a collapse. The proliferation protection levels of fresh MOX-fuel containing small additions of protactinium are evaluated for equilibrium closed cycle of a light-water reactor (LWR). Modular fast reactors cooled by lead-bismuth coolant have been mastered in conditions of operating reactors of Russian nuclear submarines. Based on this experience, the small power fast reactor for multi-purpose usage (SVBR-75/100) has been developed. Exploitation of VVER-440 as the first-tier reactor at the multi-tier transmutation system is suggested. Small Nuclear Power Plants (SNPPs) system can become experimental site for improvement of technical decisions which may be used for sustainable development large scale NE system.

South Africa: The relevant technical and economical benefit of the well known PBMR project were described.

Slovakia: Exploitation of VVER-440 as the first-tier reactor at the multi-tier transmutation system is suggested. Influence of appropriate modification of fuel management on VVER-440 fuel cycle back-end is analyzed numerically by the spectral code HELIOS. A Theoretical model is based on preliminary evaluation of VVER-440 transmutation potential. Partially closed fuel cycles of the reactor VVER-440 are characterized.

Switzerland (and France, Italy, Japan, Netherlands): Reported on the Inert Matrix Fuel developments initiative that could be deployed to utilize, reduce and dispose both weapon- and light water reactor- grade plutonium excesses. In addition to plutonium, the amounts of minor actinides are also increasing. Surpluses of these actinides have to be consequently handled in a safe, ecological and economical way.

Turkey: The primary concern of the Turkish Atomic Energy Authority for innovative nuclear reactor technologies are in the areas of: (i) resources, demand and economics, (ii) safety, (iii) environment.

Ukraine: The role of innovative technology is foreseen in the modernizations of the existing facilities.

USA: A study identified three general categories of analytical approaches with excellent potential for use in nonproliferation evaluation. Attribute analysis, scenario analysis, and two-sided methods. The goal of ANL and LANL is to develop a demonstrably effective safeguards system for advanced fuel processing technologies via targeted processes and facility modifications and the utilization of modern safeguards techniques.

Vietnam: A Pneumatic Transfer System (PTS) of the Dalat Nuclear Research Reactor (DNRR) has been playing an important role in the field of Instrumental Neutron Activation Analysis (INAA) and for the future engineering capability.

10. CONCLUDING PANEL (SESSION 7)

P. Bernard, France:

There are large discrepancies in the number of people that have access to electricity and those that don't have access. Technically, Fast Reactors are ranked first in terms of Uranium consumption sustainability and actinides management. France fully supports the global system approach developed in Generation IV International Initiative.

R. B. Duffey, Canada:

The Innovative Technologies development should be based on collaborative R&D projects (Gen IV) and international broad cooperation (IAEA-INPRO). Factors to be considered include economics, learning, proliferation resistance, material knowledge, hydrogen fuel, waste reduction and synergetic cycles. Usual disagreements in the technical path are expected.

P. Florido, Argentina:

Crisis means change and change is an opportunity. The young generation feels that the old generation is not comfortable with new solutions. The young generation was not part of the cold war. The nuclear world means for us India, Japan, Rep. of Korea, and Russia. The examples of Capston Turbine Company and Nokia Mobile Phones were given as the new paradigm. All the factors influencing the innovation like economics, environmental, etc, are a moving target that set a new paradigm for the technology to be developed.

M. Lawrence, USA:

The future of nuclear energy depends upon the transportation and primary energy uses. The main problem that will arise is to give financial support to all initiatives. Nuclear technology should face the multidisciplinary scope developed in the biological science, where physics, chemistry and biology must converge in common and coordinated efforts to give solutions. Actions in nuclear technology require to be hyperactive now to harmonize efforts and accept the risks to satisfy public requirements. The role of IAEA should be to provide a forum for Member States (developers and users), evaluate the proposed concepts against user's requirements, provide institutional innovations, and disseminate the information.

S.Saito, Japan:

Japan is pursuing a broad programme on an innovative system that includes HTGR, FBR and ADS. Japan offers the HTGR and Monju Reactor for international cooperation.

B. Wahlstrom, Finland:

Nuclear energy needs to continue growing now. We cannot wait too long because if nuclear energy does not satisfy the requirements of today there will not be a future. It is necessary to close the gap between the user requirements and technology that is offered. The new ideas are welcome now and not for a long-range scenario because in twenty years nuclear energy may be phased out of the energy basket.

V.M. Mourogov, IAEA:

Member States in the General Conference of 2001 encouraged the Secretariat of the IAEA to pursue a continuing action for developing innovative technologies for nuclear fuel cycle and nuclear power. INPRO was a response to this request and the results achieved until now force us to be more proactive in this respect. Innovative technology is an answer to a real technology gap we foresee in the future if developing countries are to have success in achieving energy consumption close to that of developed countries.

11. CONFERENCE CHAIRPERSON'S SUMMARY

All relevant aspects of innovative technologies for nuclear fuel cycles and nuclear power were discussed in an open, frank and objective manner and the conclusions are briefly summarized below.

No large increase in the use of nuclear energy is foreseen in the near and medium term, but is likely in the long term if developing country per-capita electricity consumption reaches that of the developed world. The nuclear sector including regulators view an increased use of nuclear energy as the solution for global sustainable energy needs considering that significant reductions in CO_2 emissions would be required. Although the current nuclear technology is considered to have matured as an industry, innovation is foreseen for further improvement of safety, economy, sustainability, non-proliferation, etc. On the other hand, the general public, politicians and environmental Non Governmental Organizations (NGO's) in many countries view nuclear specialists with distrust. In their view nuclear energy is not needed in the short and medium term and likely not also in the long term. Innovative fuel cycles and nuclear power technologies have to achieve inherent safety, proliferation resistance, foolproof measures against terrorist acts and sabotage, etc., even for being considered as an option.

Thus there is a gap to be bridged if the potential benefits of nuclear energy are to be realized for peace and prosperity of humanity. Technical measures such as well-defined user-requirements, improved design concepts and applications in addition to electricity generation, have to be developed. Communication has to be substantially improved both within the nuclear community and with the public and society at large. Apart from achieving acceptable economic targets in terms of cost per installed kilowatt and investment cost, it would be necessary to seek appropriate solutions for improving the investment attractiveness of nuclear plants in developing countries.

The conference succeeded in bringing together top managers, policy makers and specialists from developed and developing countries as well as representatives of R & D activities in MS and international projects. There was a broad agreement amongst the participants that international collaboration in general and the collaboration especially between Gen IV and INPRO initiatives should be improved and substantially expanded. The IAEA is expected by all to play a key role in coordinating international efforts to develop innovative technologies.

OPENING ADDRESSES AND

PANEL ON

CONFERENCE THEME

(Session 1)

Chairperson

L. ECHÁVARRI
OECD/NEA

NUCLEAR ENERGY: THE ROLE OF INNOVATION

Mohamed ElBaradei

Director General

INTERNATIONAL ATOMIC ENERGY AGENCY

Let me begin by expressing my strong support for this conference on innovative technologies for nuclear fuel cycles and nuclear power. As you well know, nuclear power is a significant contributor to the global supply of electricity, and continues to be the only source that can provide electricity on a large scale with a comparatively minimal impact on the environment. But it is equally evident that, despite decades of experience with this technology, nuclear power today remains mainly in a holding position, with its future somewhat uncertain.

For a number of years, I have been stressing that the future of nuclear power will depend on a number of factors, including: vigilance in ensuring the continued safety of operations at nuclear facilities; the development and demonstration of clear national and international strategies for the disposal of high level radioactive waste; the ability to compete economically with other energy sources; and successful outreach to civil society.

But any major future expansion in the use of nuclear power will depend heavily one additional factor: the innovation in reactor and fuel cycle technology that is the focus of this conference — innovation that successfully maximizes the benefits of nuclear power while minimizing the associated concerns. The nuclear power industry and the nuclear community must demonstrate their ability to adapt, both technologically and in other ways, to the evolving energy needs and concerns of the marketplace. I would like to share with you a few of my ideas and beliefs about how to make this innovation successful.

GLOBAL DEMAND

First, the scope of our vision for the future of nuclear power must be *global*. While we often point out that nuclear power currently provides about 16% of global electricity, we note less often that some 83% of nuclear capacity is concentrated in industrialized countries. By contrast, over 2 billion people in developing countries remain without reliable energy supplies, a major factor in their aspirations for social and economic development. Projections of future energy demand — such as that provided in the Special Report on Emission Scenarios by the Intergovernmental Panel on Climate Change — show enormous energy growth for at least the next 60 years, but clearly concentrate that growth in the developing world.

If nuclear power is to play a major role in meeting this demand for additional energy, it will require innovative approaches — both technological and otherwise — to match the needs of users not only in industrialized countries but also in the developing world. This will include consideration of grid capacities, infrastructure requirements, and other technological aspects, but it will also require foresight in building the necessary social capacities — the scientific and technical skill base, the legal and regulatory frameworks, and the public acceptance — that are necessary to make nuclear power a viable option.

RESPONSIVENESS TO CONCERNS AND DEVELOPMENTS

Secondly, innovation must be responsive to concerns that remain about nuclear power, and should be 'smart' in taking into account new developments and expected future trends. For example, innovation should ensure that new reactor and fuel cycle technologies incorporate inherent safety features, proliferation resistant characteristics, and reduced generation of waste. Consideration should be given to physical protection and other characteristics that will reduce the vulnerability of nuclear facilities and materials to theft, sabotage, and terrorist acts.

Awareness of needs other than electricity generation can help to make the nuclear contribution more substantial. In smaller communities, this could include using nuclear technology for the co-generation of electricity and district heating. In the transportation sector, the use of nuclear energy to produce hydrogen could play a strong role in supporting the expanded use of fuel cells. And the nuclear desalination of seawater, if demonstrated as a feasible and economically competitive technology, could help to address the increasing global challenge regarding the availability of freshwater. I would note that the Agency has provided assistance to two nuclear desalination projects that are well underway — the Indian project at Kalpakkam and the Pakistani project at Karachi — as well as a number of international studies for desalination plants in other locations.

THE VALUE OF COLLABORATION

Third, I firmly believe that these innovation efforts should be co-operative and collaborative in nature. The most important outcome of this collaboration may be, as I have already suggested, a better understanding of user needs and requirements worldwide. But international collaboration will also make our efforts more efficient and effective, by promoting technical information exchange, sharing safety and non-proliferation insights, and leveraging research dollars. Clearly, at some point in the development of a given technology, collaboration must give way to commercial competition; however, even after these new technologies become competitive, collaboration will continue to be beneficial for new designs with enhanced features to reduce costs and improve safety and security.

The IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was developed with precisely this objective in mind — to engender the broadest possible international collaboration, to enable the scientific and technological innovation that would ensure that nuclear energy remains a viable option for future generations. INPRO recently completed its efforts to define user requirements related to economics, safety, proliferation resistance and the environment, bringing Phase 1A of the project to a close. The INPRO Steering Committee last month approved the Phase 1A report, and made a number of recommendations for moving forward, including the pursuit of case studies that would enable Member States and independent analysts to apply INPRO methodology in specific situations. But the Committee also recommended — as I have been encouraging for some time — that INPRO strengthen its co-operation with other initiatives on innovative nuclear energy systems, including the US initiated Generation IV project. The results of INPRO's efforts to date will presented later in this conference, as well as the results of Generation IV and other projects. It is my hope that these presentations will make evident more opportunities for collaboration among these projects — collaboration that will be of mutual benefit to all concerned.

BEYOND TECHNOLOGICAL INNOVATION

Fourth, I would emphasize that our innovation efforts must be more than purely technical. The evaluation of new design aspects by the nuclear industry should be accompanied, throughout the nuclear community, by a re-evaluation of technology policy issues. These issues play a significant role in economic costs, investor confidence and public acceptance of nuclear technology. A high level of confidence must be achieved in the reliability of construction schedules, licensing review procedures, regulatory oversight, liability issues and other factors that affect the cost and efficiency of nuclear facility design, construction, startup, operation and maintenance.

Another non-technological aspect of innovation relates to our approach to societal outreach. Decision members, public interest groups, and the public at large must be engaged in a fair evaluation of the relative merits of different energy options. Improved public understanding of radiation and nuclear issues is essential if we are to create a more mature awareness of the comparative risks and benefits of different energy sources, the impact of each option on sustainable development, and the range of societal benefits provided through nuclear application. As recent world events have demonstrated, the public must also be given credible assurance that nuclear technology and materials will be used exclusively for peaceful purposes. And innovative approaches will be essential to attract the necessary new generation of talented scientists, engineers and technicians.

CONCLUSION: TIMELINESS OF THE CONFERENCE

In my view, this conference comes at a pivotal point in the history of nuclear energy. Progress has been made recently on many fronts related to nuclear power, including waste disposal, license extension, and safety and security upgrades. Many countries are engaged in innovation projects: in fact, some 20 or 30 innovative designs are currently under development, with all of the principal reactor concepts — water, liquid metal, or gas cooled — as well as accelerator driven systems being addressed in one or more projects. And while most of the current expansion in nuclear energy is taking place in East and South Asia, recent years have witnessed statements and actions in North America, Europe and elsewhere that support a renewed consideration of the merits of nuclear power. On the other hand, many countries continue to either reject or express strong reservations about the nuclear option — primarily due to concerns related to waste, safety and security.

The 21st century promises to deliver the most competitive, globalized markets — and the most rapid pace of technological change — in human history. In the next 50 years, we are also likely to witness the greatest expansion of energy use ever known, particularly in developing countries. For nuclear technology to make a substantial contribution to energy supplies, innovation is essential — innovation that is global in scope, responsive to concerns, and collaborative in its approach.

I encourage your enthusiastic participation in this conference as you seek to address many of these issues in the next few days, and I look forward to the results of your discussions.

PERSPECTIVE OF A DEVELOPING COUNTRY WITH EXPANDING NUCLEAR POWER PROGRAMME

A. KAKODKAR

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Abstract. At the present stage of development no single energy resource or technology constitutes a panacea to address all issues. Therefore, it is necessary that all low-carbon and non-carbon emitting resources become an integral part of an energy mix – as diversified as possible – to ensure energy security to the world during the present century. Available sources are low carbon fossil fuels, renewables and nuclear energy and all these should be subject of increased level of research, development, demonstration and deployment. It is in this context, that the Agency has an important role to play. Growth of nuclear energy in the developing countries is a matter of global interest in view of its potential to protect the earth from irreversible climate changes. The present nuclear energy technologies need to be examined and barriers – real or perceived – to their deployment overcome through technological solutions. Concerns with regard to safety, restrictive regimes arising out of international politics, and investment risks due to long gestation periods, evolving liability regimes and uncertain regulatory regimes need to be addressed through innovative solutions. IAEA with its broadest membership base is an indispensable platform for addressing these concerns through a technological approach

1. INTRODUCTION

The world's population crossed the 6 billion mark in the year 1999. Most current estimates suggest that around 2 billion people will be added over the next 30 years with another billion in the following 20 years. Virtually all this increase will be in the developing countries with the bulk in urban areas. The core challenge for development is to ensure availability of productive work opportunities and a better quality of life for all these people [1]. Two aspects are very important, viz., quality of life should be above a minimum threshold and there should be equitable opportunities for all. At present, however, inequality is widening. The average income in the richest 20 countries is now 37 times that in the poorest 20 and this ratio has doubled in the past 40 years. Inequalities can give rise to conflicts and therefore, it is necessary to address development concerns of all nations.

Energy is the engine for the growth. It multiplies human labour and increases productivity in agriculture, industry as well as services. Thus, easier access to energy in the developing world holds the key to bridging the widening inequality. However, the inequality seen in income level is also seen in per capita energy consumption. Statistics published by the IAEA [2] indicate that per capita energy consumption in North America in 2001 was 343 GJ and it is expected to grow to 346 – 387 GJ by 2020. Per capita energy consumption in Africa is expected to change from 27 GJ in 2001 to 26 - 32 GJ in 2020; in the Middle East and South Asia from 25 GJ in 2001 to 30 - 38 GJ in 2020. These forecasts do not indicate any perceptible improvement in the inequality. With sustainability issues staring at us, the above situation can be corrected only if the energy supply becomes abundant and within the reach of all. Only power of the atom can in principle realize this.

Rapid developments in nuclear power technology in sixties and seventies have demonstrated practical feasibility of large-scale role that nuclear power can play in meeting the energy challenge. However, this deployment has largely been restricted to the industrialized world which is by and large in a stable mode in as far as energy demand is concerned. The desperate need for growth in energy availability exists in the developing world, both because the per capita energy consumption needs to be taken to a much higher level and also because of the growth in population which would stabilize only when survival no longer remains an issue and there is a general feeling about an assured reasonable quality of life.

As the developing world tries to meet the energy needs of its growing population and support its development aspirations, the global energy consumption would double over the next three decades and will rise further subsequently. It is clear to me that without a central role for nuclear power this could lead to a catastrophe both in terms of sustainability of energy resources with enhanced level of conflicts to grab residual resources and even more importantly in terms of global climate. As we move forward in time, the crucial importance of nuclear power would be increasingly felt not only for supporting economic growth but also for some basic human needs such as availability of clean air and water. In fact, the day is not far off when we would need to view nuclear energy as not just a source of electricity but a primary energy source which could assure our sustainable future.

Developments in science and technology have led to the improvement in quality of human life. Although new problems have arisen in the process, these have in fact been solved by further developments in science and technology. For example, today we can justifiably be proud of increased longevity realized through emphasis on health and nutrition programmes. The increased demand for food as a result has been met through advances in agriculture. Looking back to the 1950s and 1960s, it was then feared that the developing countries - particularly China, India and Indonesia - would not be able to feed their rapidly growing populations. Thanks to the green revolution in agriculture, the doomsday scenarios of famine and starvation in these, the most populous, developing countries were proved wrong. [1]. Given the inevitable role nuclear power is required to play in years to come, there is thus a strong need to examine further technological solutions that need to be brought about to overcome barriers that exist in its large scale deployment in the developing world.

Looking from the perspective of a developing country like India, development of nuclear energy based on a closed cycle approach enabling fuller use of uranium and thorium is inevitable for development aspirations of over a billion people. The electricity generation in the fiscal year 2002-03 was about 531 billion kWhr from electric utilities [3] and additional about 127 billion kWhr were generated by the captive power plants [4]. On per capita basis, this works out to 620 kWhr per year (Table I). India's GDP in recent years has been increasing at 6% per year. Development aspirations of its people demand that growth at this or even at a higher rate be sustained for a long enough time. Indian government has taken several steps to realize these aspirations. These include policy initiatives as well as planning and launching of projects aimed at improving the energy, transport and water infrastructure in the country. Examples include the ongoing project to build a network of national highways, setting up a task force to prepare a blueprint for linking major rivers in the country to solve the problem of recurring floods in some parts and drought in the other parts of the country and ongoing reforms in the electricity sector with electricity bill - 2003 having been passed by parliament only a few weeks back. Several other initiatives have been taken such as new telecom policy which has already resulted in rapid growth of telcom infrastructure in the country. All these are steps towards achieving an average annual growth of 8% during the ongoing 10^{th} five-year plan (April 2002 to March 2007).

Table-I. India – Some statistics

Population as on 31.03.2003	~1.062 billion
Installed capacity as on 31.03.2002	107,972 MWe (Utilities)
	~29,000 MWe (Captive power plants)
Gross electricity generation in 2002-3003	~127,020 (Captive power plants)
	531,607 million kWhr (Utilities)
Per capita generation	~620 kWhr/year

In terms of electricity generation, India would have to plan to reach at least a modest target of electricity generation of 5000 kWhr per year per capita. India's population could rise to 1.5 billion by the year 2050. This would call for a total electricity generation of about 7500 billion kWhr per year. This is an order of magnitude higher than the generation in the fiscal year 2002-03 and calls for a careful examination of all issues related to sustainability including abundance of 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.

This kind of situation is true for several other countries on a growth path and at the present stage of development no single energy resource or technology constitutes a panacea to address all issues. Therefore, it is necessary that all low-carbon and non-carbon emitting resources become an integral part of an energy mix – as diversified as possible – to ensure energy security to the world during the present century. Available sources are low carbon fossil fuels, renewables and nuclear energy and all these should be subject of increased level of research, development, demonstration and deployment.

Let us examine the fuel resource situation in India (Table II). Estimates by us in the Department of Atomic Energy (DAE) and also by other agencies in the country indicate that we will have difficulties with regard to availability of coal by the middle of the present century. In addition, coal based stations are likely to pose serious problems in future arising out of transport of large quantities of coal across the country and environmental problems related to disposal of ash and emission of greenhouse gases and acid gases. Our oil and natural gas reserves are very modest and we are importing a very significant share of our requirements, which constitutes a major share of our overall imports. Hydro-potential is renewable and must be exploited to the maximum. Exploitation of hydro-resources is handicapped by issues like displacement of people. Non-conventional sources like solar, biomass and wind will play useful roles, but at the present level of technology development they can only complement electricity generation by base load stations based on fossil, hydro or nuclear plants.

Table II. India's Energy resource position

Resource	Amount	Potential (GWe-yr)
Coal	221 billion tonne (Total)	44,000
Oil	0.75 billion tonne	300
Natural Gas	692 billion Cu.m.	250
Hydro	84 GW at 60% PLF	84 GW at 60% PLF
Uranium	78,000 tonne metal	In PHWRs – 420
		In FBRs – 54,000
Thorium	518,000 tonne metal	In Breeders – 358,000
Non-conventional		
Wind	20	
Small Hydro	10	
Total Solar Insolation	600,000	
Ocean thermal, Sea wave & Tidal	79	
Assumptions for potential calculation	For Coal, Oil and Gas: Complete source is used for electricity generation with thermal efficiency, η = 30% and calorific value for Coal = 5000 kcal/kg, Oil = 10,200 kcal/kg & Gas = 9150 kcal m^3 For Nuclear: Fuel burn up in PHWRs = 6700 MWD per tonne & η=29% FBRs can use 60% uranium with η= 42%. Breeders can use 60% thorium and η= 42%	

Our uranium deposits are limited, while thorium deposits are large. To maximize energy potential of available nuclear resources, a closed fuel cycle involving reprocessing of spent fuel to recycle plutonium and uranium-238 has to be pursued. Besides recovering valuable fissile and fertile materials, reprocessing helps to sort out the wastes according to their activity levels and their decay period thereby assisting waste disposal and minimizing environmental impact. The development and experience in closed uranium fuel cycle would soon need to be expanded to cover thorium fuel cycle to ensure long-term energy security for the country.

Closed cycle has the capability to virtually de-couple energy supply from resource related constraints for generations to come.

Indigenously developed Pressurized Heavy Water Reactors (PHWRs) and associated fuel cycle facilities are being established to meet current electricity needs and future fuel needs. At present, we have 12 such reactors in operation and six under construction, which include larger indigenously designed and developed 540 MWe units under construction at Tarapur. The designs of these reactors have progressively evolved taking into account the needs for indigenisation, our own operating experience, operating experience in PHWRs outside the country and progressive evolution of enhanced safety features. We are self sufficient in all aspects of PHWR technology. As we gain experience and master various aspects of the nuclear technology, performance of our plants is also improving. Average capacity factor of our plants have steadily risen from 60% in 1995-96, to 90% in the year 2002-03 (Figure 1). Nuclear power plants have so far produced about 200 billion units. We have accumulated about 200 reactor-years of operational experience free of any serious incident involving release of radioactivity to the environment.

We started FBR programme with the setting up of a Fast Breeder Test Reactor at Kalpakkam. This reactor, operating with indigenously developed mixed uranium-plutonium carbide fuel has achieved all its technology objectives. Based on the experience gained with this reactor and with active cooperation of academia and industry, detailed design and technology development of the 500 MWe Prototype Fast Breeder Reactor (PFBR) has been completed. Pre-project activities for this project have already commenced at Kalpakkam near Chennai. Overall, we plan to have an installed nuclear capacity of about 20,000 MW by the year 2020 (Table III).

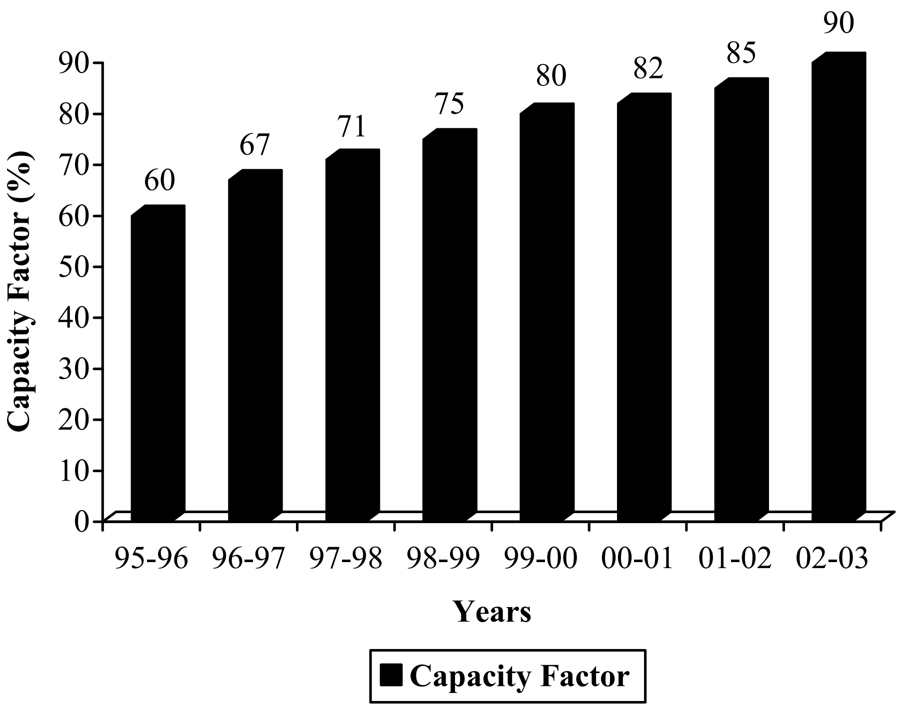

FIG. 1. Average capacity factor of nuclear power plants in India.

Table III. Nuclear power plants – Present status and future plans

Plants under operation	MWe
14 reactors at 6 sites viz.,	
— Tarapur, Rawatbhata, Kalpakkam	2720
— Narora, Kakrapar and Kaiga	
Plants under construction	
— 2x540 PHWR at Tarapur	
— 2x220 PHWR at Kaiga	3960
— 2x220 PHWR at Rawatbhata	
— 2x1000 VVERat Kudankulam	
Plants likely to commence construction shortly	
1x500 PFBR at Kalpakkam	500
Future Plans	13740
Total	**20920**

Timely implementation of a programme for thorium utilisation is very crucial for us to meet the increasing energy demands in the country. A small beginning has already been made by introducing thorium in a limited way in research reactors and in PHWRs. With sustained efforts over the past several years, we have small-scale experience over the entire thorium fuel cycle. A research reactor KAMINI is operating in Kalpakkam based on Uranium-233 fuel, which is derived from thorium. This fuel was bred, reprocessed and fabricated indigenously. An Advanced Heavy Water Reactor (AHWR) has now been developed at Bhabha Atomic Research Centre (BARC) to expedite transition to thorium based systems. The reactor physics design of AHWR is tuned to generate about 65% power from thorium. The design incorporates several advanced safety features. The detailed project report of this reactor has been prepared and is undergoing a structured peer review before we launch its construction in the financial year 2004-2005. AHWR design has been carried out with a futuristic vision. The design incorporates several passive safety features enhancing its operator forgiving characteristics. Being a thorium system, the long lived waste in the form of minor actinide is expected to be much smaller. We have agreed for this design to be a case study under the INPRO Programme.

As a further step towards self sustained thorium utilization with a potential for growth, a road map for the development of Accelerator Driven System (ADS) has been prepared. Development of such a system offers the promise of shorter doubling time with thorium-uranium-233 systems, incineration of long-lived actinides and fission products. ADS alongwith thorium-uranium 233 reactors and fuel cycle have the potential to provide a robust eco-friendly technology base to large-scale thorium utilization. As a first step towards realization of ADS, we are launching development of proton accelerator in the 10th five-year plan.

It is worthwhile to recognize the importance of high calorific value of nuclear fuel. Nuclear fuel contains energy in a concentrated form thus requiring much less tonnage for fuel to be transported or stored. In the overall cost of electricity generated from nuclear fuel, the cost of fuel is a much smaller fraction as compared to the other components. This can be seen by the data in the Table IV. One may dispute the accuracy of these numbers, but one cannot dispute the order of magnitude difference between the characteristics of nuclear fuel and other fuels. Thus, if only the capital cost of setting up nuclear reactors can be brought down substantially, the nuclear energy would become an abundant and cheap energy source. Today we are already building our nuclear power stations at an overnight cost of around $1100 per kWe. With development of newer technologies we expect this to go down further.

Studies carried out by us in DAE indicate that the magnitude of installed electricity generation capacity in India by the middle of this century has to be about 1300 GWe in order to generate 7500 billion kWhr per year. Sustainability as well as ecology issues would demand that a significant fraction should come from nuclear sources. R&D efforts in India are being planned to meet this goal. Adoption of a closed fuel cycle would reduce waste volume and future development of technologies for partitioning and transmutation would make high-level waste storage a short-term issue. There is significant knowledge base in most of these areas. The need of the hour is to convert this knowledge base with augmentations wherever necessary into technologies packaged in a manner that address all issues that constitute a barrier to the large scale adoption of nuclear energy.

Similar projections can be made with regard to world electricity demand. It is projected to grow by 2.4% per year from 2000 to 2030, thereby doubling electricity consumption over this period [5]. In developing countries, it will expand by 4.1% per year. As a result of worldwide increase in energy demand, carbon-dioxide emissions will increase by 1.8% per year from 2000 to 2030. Two-thirds of the increase will come from developing countries. The world's fossil fuel based energy resources are adequate to meet the increased demand over the next three decades, but it cannot be said for all times to come. Total fossil fuel supply can rise only by a limited amount in the face of continuously rising energy demand in the world. The energy gap has to be filled by nuclear power as the renewables can be expected to play only a marginal role. John Ritch [6] predicts addition of 3000 to 8000 one Gigawatt reactors over the next 50 years. Most of the demand for increase in installed nuclear generation capacity will be from the developing world. Thus we have a situation that likely demand in expansion of nuclear generating capacity is in countries which may or may not be 'nuclear capable'.

Table IV. Fuel cost at Indian port

Fuel	Rs/Tonne	Billion US $/EJ
Naphtha	13,470	5.86
L.N.G.	12,500	5.80
Coal	2,346	1.67
Natural Uranium (U_3O_8)	11,00,000	0.04 (at international market prices)

EJ = 10^{18} Joules

It is in this context, that the Agency has an important role to play. Growth of nuclear energy in the developing countries is a matter of global interest in view of its potential to protect the earth from irreversible climate changes. The present nuclear energy technologies need to be examined and barriers – real or perceived – to their deployment overcome through technological solutions. Concerns with regard to safety, restrictive regimes arising out of international politics, and investment risks due to long gestation periods, evolving liability regimes and uncertain regulatory regimes need to be addressed through innovative solutions. Special needs of developing countries as a result of lower economic strength and inferior technological infrastructure have to be understood and technologies appropriate to address the needs have to be developed so as to ensure that the energy supply infrastructure is not vulnerable to commercial interests. Nuclear energy is based more on knowledge, less on materials, than most others, and therefore, requires expertise in several disciplines of science and technology. This expertise has to be acquired through painstaking efforts and for the spread of nuclear technology, conventional technology transfer models can work only if they are accompanied by strong human resource development component, a prerequisite for technology assimilation.

IAEA with its broadest membership base is an indispensable platform for synergising these efforts. Since the INPRO programme aims to evolve a technological approach to addressing economics, safety, proliferation resistance and waste management challenges, it also represents a more fundamental way to make IAEA activities in safety and safeguards more effective and manageable even in a future scenario of large scale growth in deployment of nuclear power. Linkages with other national and international programmes such as GIF should only strengthen and speed up the INPRO approach.

2. ACKNOWLEDGEMENT

Contribution of Dr. R.B. Grover, Director, Strategic Planning Group, DAE in putting together this presentation are gratefully acknowledged.

3. REFERENCES

[1] Sustainable Development in a Dynamic World – World Development Report –2003, World Bank.
[2] Energy, Electricity and Nuclear Power Estimates for the Period up to 2020, Reference Data Series No.1, July 2002, IAEA.
[3] Personal Communication, Central Electricity Authority, May 2003.
[4] PowerLine, November 2001 gives estimates of captive power installed capacity in India and these have been extrapolated based on data given in PowerLine, December 2002. A capacity factor of 50% has been assumed.
[5] World Energy Outlook – 2002 – Highlights, International Energy Agency, Paris.
[6] John Ritch, "Nuclear Energy at a Moment of Truth", IAEA Bulletin, Vol. 44, No. 2, 2002

INNOVATIVE NUCLEAR FUEL CYCLES FOR THE FUTURE: *Why do we pursue nuclear fuel recycling?*

- Expectations towards fast reactors -

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Abstract. Modern civilization was established in the 20th century based on the mass consumption of natural resources. In the twenty first century, the world heads for a recycling-oriented civilization. Nuclear energy is a technology suitable for this age. Current nuclear fuel cycle technology aims for the development of fast reactor cycle. In Japan, Monju plays a key role for fast reactor fuel cycle technology development. In pursuit of this goal, Japan Atomic Energy Research Institute (JAERI) and Japan Nuclear Cycle Development Institute (JNC), the two key players in the development of nuclear technology will be merged. Considering that people in the world are looking forward to the growth and development of recycling-oriented society in the 21st century and that the expectation in Japan is so high for nuclear energy development, it is hoped that a high level of participation in the joint development and cooperation for Monju as well as HTTR from international community would make these projects the front runner of the worldwide efforts.

1. IN THE 20TH CENTURY THE MODERN CIVILIZATION WAS ESTABLISHED BASED ON THE MASS CONSUMPTION OF NATURAL RESOURCES

In the 20th century, we have sought convenience and established affluent modern civilization based on the progress of science and technology since industrial revolution. The primary energy sources maintaining modern civilization are fossil fuel accumulated over the hundreds of millions or billions of years.

We have used a huge amount of fossil energy in a short period of time and succeeded in the rapid growth of modern civilization. Also we have disposed a huge amount of the generated C02 and other chemical materials harmful to the environment and human being. As a result, the environmental pollution and global environmental problems became obvious. The environment of the earth seemed infinitely great to mankind in the past. However, we clearly recognized the limit of the environment by emergence of global environmental problems.

On the other hand, the limitation of fossil energy has been pointed out for quite a long time. The future growth of energy consumption with the modernization in Asia may bring about the short supply in near future.

Nuclear energy has a technical advantage that high fuel energy density makes storage easy, and it does not produce green house effect gas, carbon dioxide, in the process of power generation. So it is effective measures against global warming. Compared with other energy sources, the amount of waste produced per same unit of energy use is very small and it does not require vast space for storage or disposal.

Because of these merits, many advanced countries started to introduce nuclear power plants in the last half of 1960s. At present, more than 400 LWR reactors are in operation and providing a major part of energy supply. The success of LWR in the world was strongly supported by the success of the front end of nuclear fuel cycle, i.e. enrichment of U235.The mankind realized to produce the natural uranium of 20 billion years ago, the fuel of a natural fission reactor, which is similar to LWR. The second step for nuclear fuel cycle is to recycle fuel. In Europe, MOX has been used in light water reactors over 20 years. That is, the nuclear energy use has entered into a maturing stage.

2. IN THE TWENTY FIRST CENTURY, THE WORLD HEADS FOR A RECYCLING-ORIENTED CIVILIZATION. NUCLEAR ENERGY IS A TECHNOLOGY SUITABLE FOR THIS AGE

In recent years, priority is put on the policy of easing environmental effect by the fossil fuel consumption rather than resources point of view. The Kyoto protocol shall be the international consensus for the environmental protection. The limitations of resources and environmental capacity request us to make an end to the mass consumption and mass waste-disposal and build a recycling – oriented society. For this purpose, in the 21st century, the science and technology is needed not only for the utilization of natural resources but also for the enhancement of the efficient use of resource and reduction of the waste.

In once-through nuclear fuel cycle with light-water reactors, only 0.5 percent of natural uranium can be utilized, leaving fissile uranium and fissile plutonium in the spent fuel. From the viewpoint of spent fuel's radioactive toxicity and its duration, there is great advantage in the recycle of nuclear fuel, compared with direct disposal of spent fuel. In addition, future possibility of diversion of fissile plutonium for nuclear weapons is of great concern. In the result, once-through nuclear fuel cycle with light water reactors is not considered as appropriate and reasonable as to be recommended from the viewpoints of effective utilization of natural resources, environmental safety and nuclear proliferation resistance. The transition from once-through nuclear fuel cycle to closed one meets the current trend of recycling-oriented society.

From a global point of view, nuclear fuel cycle has been promoted in France and United Kingdom as well as in Japan. In the United States, commercial nuclear fuel reprocessing was abandoned by the Carter Administration, and the direct disposal of spent fuel, which allows future retrieval of the fuel, has been considered. However, not only once-through nuclear fuel cycle, but also transmutation technologies for plutonium and transuranic elements with accelerators and reactors are recently explored within the frameworks of NERI, GEN-IV and AFCI.

3. CURRENT DEVELOPMENT OF THE NUCLEAR FUEL CYCLE TECHNOLOGY AIMS FOR THE DEVELOPMENT OF FAST REACTOR CYCLE

For the stable supply of energy and protection of the environment, what is expected in the 21st century must be the closed nuclear fuel cycle system which is proliferation -resistant, with safety as its first priority. Through our R&D efforts we have to establish an advanced nuclear proliferation-resistant fuel cycle system in which plutonium does neither exist as separated element nor its existence have any significance, rather plutonium is co-dealt with uranium and other transuranic elements.

Fast reactor in which fast neutron is playing a major role is placed at the center of such system. One can find R&D of such advanced cycle system has been under way in the international nuclear arena. France and Russia are also active in the development of fast reactors while France's R&D efforts in this area focus more on the gas-cooled fast reactor. We expect this trend of the development of the innovative nuclear system to gain momentum and international solidarity in this area to be strengthened.

4. MONJU PLAYS A KEY ROLE FOR OUR FAST REACTOR FUEL CYCLE TECHNOLOGY DEVELOPMENT

Atomic Energy Commission of Japan has pointed out the importance of the development of nuclear fuel cycle from the perspective of efficient utilization of resources and reduction in the burden to the environment. It is our responsibility to emphasize that making use of nuclear energy only for peaceful purposes, and securing the nuclear non-proliferation, will bring about the future to the nuclear energy development. At present, we are making a lot of efforts firstly for the completion of LWR fuel cycle, i.e. operation of Rokkasyo-mura reprocessing plant, MOX fuel utilization in light water reactors as many as 1618 plants and site selection for HLW to keep our international commitment that we don't possess any excess plutonium.

On the R&D front we have sought the establishment of advanced nuclear fuel cycle, carrying out the R&D for fast reactor and particle accelerator.

Monju, positioned as the core of our long-standing R&D efforts for the fast reactor fuel cycle, has stopped its operation since the sodium leak incident in December 1995. However, the importance of Monju has not been changed. When it resumes the operation, it will be utilized for the demonstration of its reliability as the power plant and the establishment of the sodium treatment technology through its operational experience.

Japan has pursued the cooperation with France and Russia for the R&D of fast reactors and related fuel cycle while we are in a process of discussion for more active cooperation with the U.S. Department of Energy. With the progress of globalization, I believe we should pursue efficient and optimal international cooperation by sharing resources, not the isolated effort by the individual country. We need a long-term effort and financial cost for the development of fast reactor system which benefits future generation. International collaboration and mutual support are necessary to obtain the public understanding in each country as well as in the international community.

We consider Monju as a facility invaluable for international cooperation. We intend to make this reactor open to the international community. We hope the cooperation from other countries. I

5. JAPAN ATOMIC ENERGY RESEARCH INSTITUTE (JAERI) AND JAPAN NUCLEAR CYCLE DEVELOPMENT INSTITUTE (JNC), THE TWO KEY PLAYERS IN THE DEVELOPMENT OF NUCLEAR TECHNOLOGY WILL BE MERGED

JAERI and JNC have played principal roles in the R&D for the fast reactor HTTR and other innovative nuclear systems in Japan. It was decided that these two institutes would be merged. This is a part of the overhaul of the public sector in which governmental departments already were reorganized and the reorganization of the government-funded corporations is under way. In our view, the creation of the new entity which covers almost all the aspects of R&D for nuclear energy is significant for the implementation of nuclear energy policy in Japan. The R&D of the new entity is wide-ranged, including nuclear fusion, accelerator and the use of radiation. However, the nuclear fission is expected to be the active and crucial field through the consolidation of both organizations' resources. This area of R&D which includes research reactors, high temperature gas-cooled reactors, fast reactors and its associated fuel cycle, and the disposal of high-level radioactive waste, must be one of the challenging issues for the new entity.

In the area of the innovative nuclear system, we intend to contribute to the international community through the development of the high temperature gas reactor, HTTR, which serves also for the hydrogen production, and the development of fast reactor cycle with Monju as its main facility. The new entity is expected to play a key role in this area. Continuous efforts will be necessary for a new entity to be internationally recognized and contribute to the international collaboration.

6. CONCLUSION

Understanding that people in the world are looking forward to the growth and development of recycling-oriented society in the 21st century and that expectation is so high for our nuclear energy development, I would like to conclude my remarks by hoping that participation in the joint development and cooperation for Monju as well as HTTR from international community, would make these projects the front runner of the worldwide efforts.

NEEDS, PROSPECTS AND CHALLENGES FOR INNOVATIONS

(Session 2)

Chairperson

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ENERGY DEMANDS AND THE NUCLEAR ROLE

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Abstract. In 2001, the time appeared to be propitious for nuclear energy with the just released energy policy in the US and an increased support for nuclear energy in American opinion polls. However the subsequent two years have not been especially good for nuclear energy due to increased concerns for safety and proliferation. The paper offers some ideas about future energy trends in order to set the context in which innovation must succeed. The drivers of energy demand are identified and the implications for different energy sources in particular, for electricity generation are analysed. It is concluded that innovation is most important to ensure that nuclear energy is a substantial contributor to a carbon-constrained, sustainable energy future.

When I had the honour to address the International Atomic Energy Agency's conference on medium and small reactors in May 2001, I congratulated the organisers on choosing a propitious time. The Bush Administration's Energy Policy had just been released, signalling an increased role for nuclear power. With the California electricity crisis fresh in people's minds, American opinion polls showed increased support for nuclear energy. Even The Economist had been prompted to feature a nuclear reactor on its front page, with the title "A new dawn for nuclear power?"

It would be fair to say that the two years since have not been especially good for nuclear energy. First we had the revelations of falsified safety records at Tepco nuclear reactors. While the concerns were more procedural than actual safety, the eventual shutting down of 17 reactors, albeit it temporary, has tarnished nuclear energy's energy security credentials. As a result of the shutdown, Japan's coal imports were pushed up 50% on the previous year and her significantly increased oil imports were one of the factors pushing up oil prices in the early months of 2003.

Then we have had the near-bankruptcy of British Energy, following the introduction in UK of a new electricity trading system which intensified competition. This was a company which at one point was a stock market darling for its robust performance in what was then seen as a "liberalised" electricity market. No matter that other generators have also had difficulties under the new trading system and that British Energy's problems were exacerbated by arbitrary cost burdens and management misjudgements, the debacle has reinforced the perception that nuclear energy cannot stand on its own feet in liberalised electricity markets.

Finally there has been the discovery that Iran has been developing enrichment technology without declaring it to the International Atomic Energy Agency, of which it is a member in good standing. No matter that technically, the Treaty which binds Iran does not prohibit research, for the average thinking citizen, it is not comforting to learn that a country can come so close to developing a key weapons technology, without infringing the non-proliferation regime or being discovered by the non-proliferation inspection teams. Arguably this is more worrying than the withdrawal of North Korea from the Nuclear Non-Proliferation Treaty, which – though a first – was always recognised as a theoretical possibility. If not a breach of the word of the Treaty, it has to be a breach of its spirit and a pointer to the need for the Additional Protocol on intrusive inspections to be a requirement for all IAEA members.

There have, of course, also been many positive developments in nuclear energy in this time - the Finnish Parliament's decision to endorse the ordering of a fifth nuclear plant for that country is clearly one of them – but there have been too many negative stories and of course they get the media coverage. Something different is needed! Hence the importance at this time of your conference on innovative nuclear technology.

As a basis for the discussion to come, I would like to offer some ideas about future energy trends in order to set the context in which innovation must succeed. You will be pleased to hear that it contains some more positive elements than my initial observations might suggest.

The World Energy Council is in the process of completing a study we have called "*Drivers of the Energy Scene*"[1] and I will to draw on this work. The study does not purport to be an energy forecast nor even an energy scenario. Rather it recognises that in developing forecasts and scenarios, it is easier to extrapolate trends than to predict discontinuities and shocks. But the one thing we can say from past experience is that the future will not be a straight-line extrapolation of present trends. So we have focused on trying to understand past discontinuities and shocks in order to gain insight into how and when they may occur in the future. For this reason I need to take you on a short journey though some energy history.

First, what have been the drivers of energy demand? We conclude that GDP (and incidentally we do our analysis using GDP at purchasing power parity) has been the fundamental driver of energy demand, with energy price as the linking mechanism shifting the rate of energy demand growth (Figure 1). GDP growth has, however, been slowing over several decades. Average annual world growth was 5% 1960-1972, 3.3% 1974-1988 and 2.8%1988-2002 (Table I).

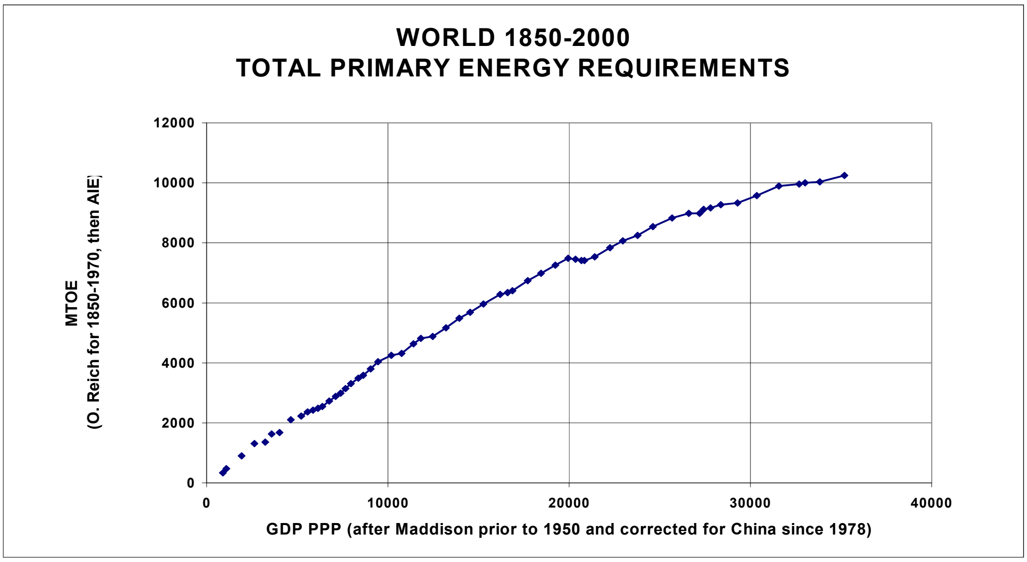

FIG. 1. GDP vs. energy demand.

[1] The WEC study, *Drivers of the Energy Scene*, which will be published in late 2003, is chaired by Majid Al-Moneef of Saudi Arabia and led by Jean-Marie Bourdaire, WEC Director of Studies.

Table I. Past GDP (PPP in T$ 1995) and GDP growth in the 1960-2002 period

Group of countries	GDP 1960	*Average growth 1960-74*	GDP 1974	*Average growth 1974-88*	GDP 1988	*Average growth 1988-02*	GDP 2002
Developed countries	5.72	*4.8%*	11.05	*2.9%*	16.49	*2.3%*	22.78
Developing countries	2.35	*5.6%*	5.05	*3.5%*	8.15	*3.6%*	13.30
Economies in transition	1.57	*5.0%*	3.11	*4.3%*	5.61	*2.9%*	8.35
World	9.64	*5.0%*	19.21	*3.3%*	30.25	*2.8%*	44.43

In fact GDP growth has been basically linear or, in other words, the percentage growth rate has been declining (Figure 2).

But if GDP drives energy demand what are the drivers of GDP growth? They can be categorised under three headings:

1. *Population growth*, which for much of the second half of the twentieth century was high. But according to UN data, the growth rate actually peaked around 1990 and has been dropping quite significantly since – far more rapidly than anticipated even 10 years ago. Whereas forecasts of a global population of 11-12 billion were quite common (including in WEC's earlier work), the UN's medium fertility scenario now sees population reaching about 9 billion in 2050 (Figure 3) and possibly plateauing shortly thereafter. There are also grounds for thinking it may not reach this high.

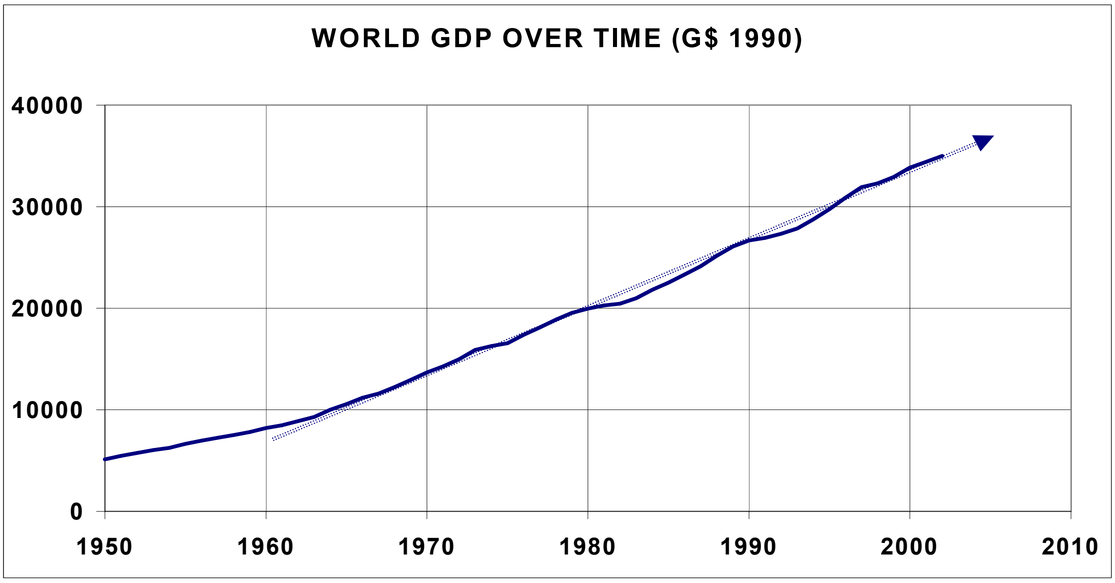

FIG. 2. World GDP over time.

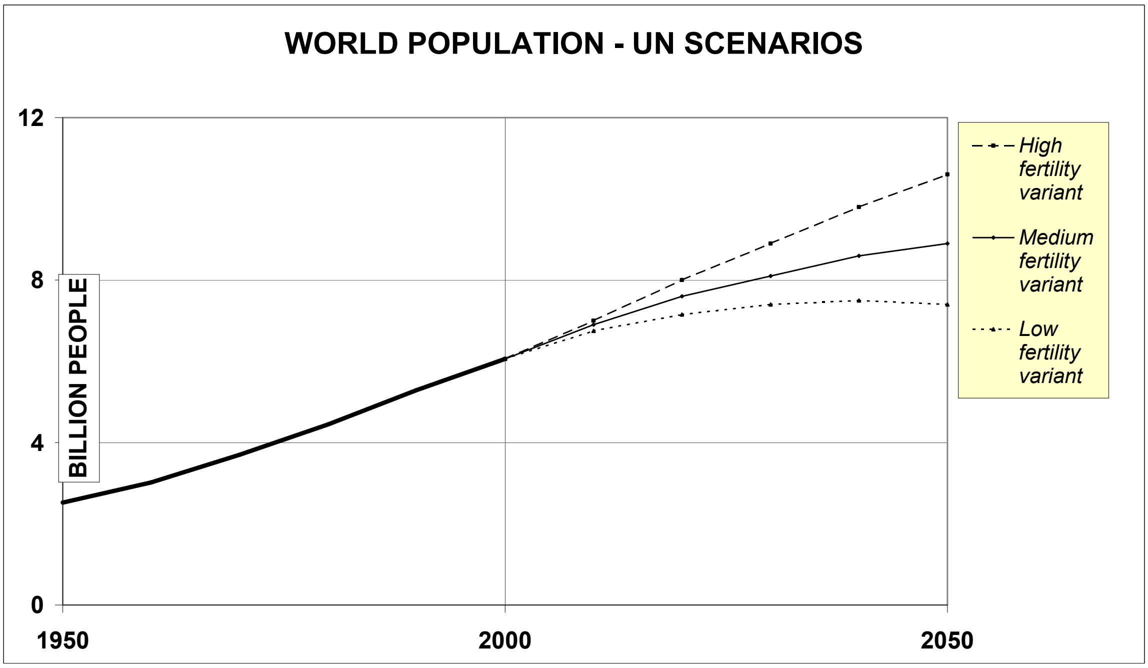

FIG. 3. World population vs. UN scenarios.

2. *Technology* is the link between population and productivity – the output of which each of us is capable. Technology-driven productivity increases were very significant in the late nineteenth century (arguably the first era of globalisation) and have also been significant in the second half of the twentieth century. We are currently in the midst of what we like to call the information technology age. While information technology has undoubtedly already done much to raise productivity further, it is still hard to say how far it will compensate for the downward effect on GDP growth of lower population growth rates. This is greatly affected by the third factor.

3. *The "d" factor*, as I call it, is a combination of attributes which together might be said to amount to "development capacity". It includes education levels, entrepreneurship, governance, institutions and infrastructure, all of which permit people to take up technology and deliver GDP growth. The "d" factor is apparently very unevenly spread and "d"factor deficits may well prove to be the key constraint on GDP growth.

Putting these observations together, it has to be said that one possibility is for GDP growth just to continue its long-term gradual decline, which unfortunately could mean that closing the gap between rich and poor will be a very long process. But does such an easing of GDP and energy demand growth present the advantage of more ample energy supply relative to demand? Again we must dive back into history.

Looking back we can see that something very fundamental happened to energy supply in 1973. The oil shock of that year is well recognised – less widely recognised is the switch which accompanied it of the fuel at the margin and the ramifications of that switch (Figure 4).

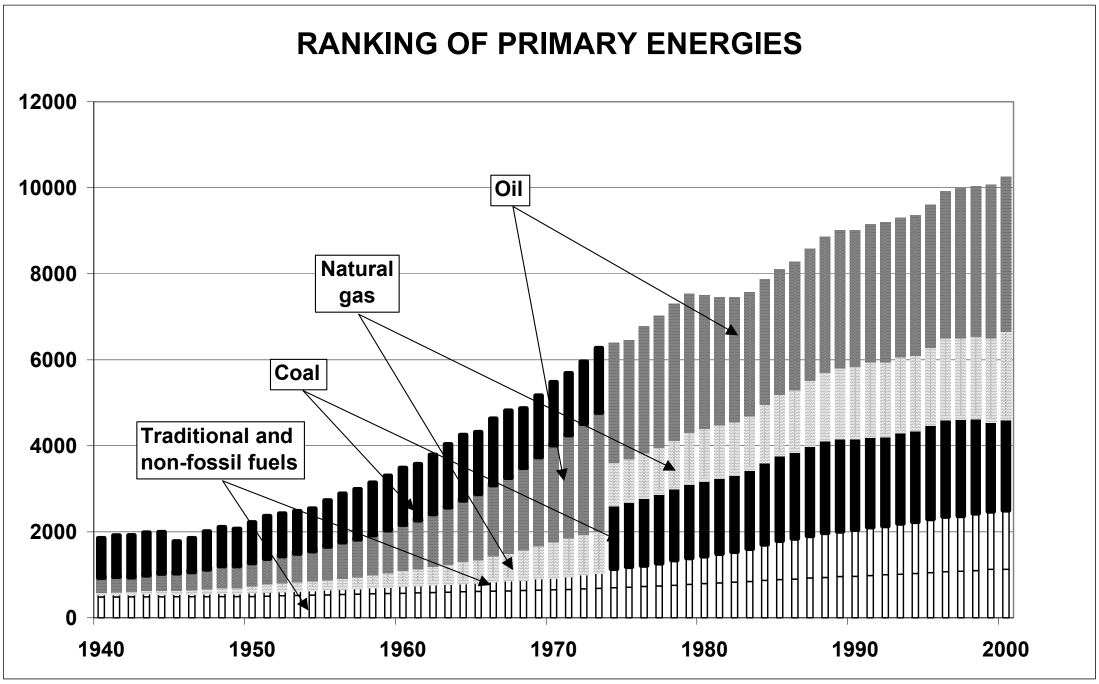

FIG. 4. Ranking of primary energies.

Up to 1973, the principal fluctuations in supply occurred with coal supply. Oil was relatively cheap and markets absorbed pretty well as much as was produced. That all changed in 1973 and oil became the swing supplier at the margin. What had happened to make this possible?

At least part of the answer, and arguably the key part, lies in the USA. US oil production has grown steadily from the beginning of the century until 1970, when quite suddenly it went into quite sharp decline (Figure 5). This delivered a shock to oil markets which permitted OPEC to muster its forces and raise prices.

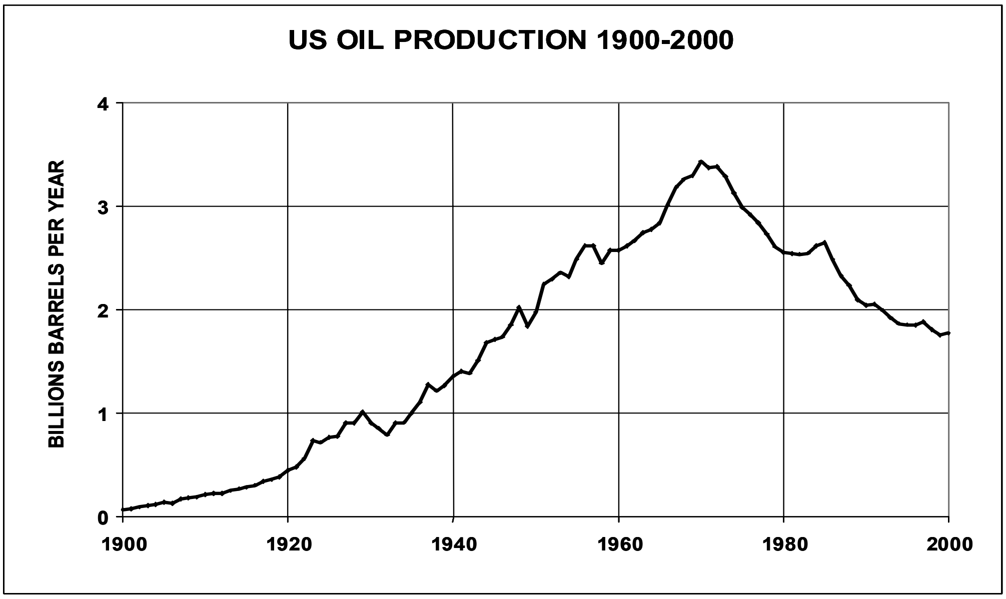

FIG. 5. US oil production.

If the peaking of production in a key market can produce such a fundamental shock to the system, it is worth considering what scope there is for repetition of such phenomena. One candidate catalyst is north American natural gas production

A predictor of the peaking of US oil production, it turned out, was the profile of oil discoveries there lagged by thirty years – the so-called Huppert curve. Superimposing a graph of north American gas discoveries on north American gas production also shows a strong correlation, but with a 20 year lag. The disquieting aspect of this is the sharply declining discovery rate from about 1978 to 1990 (Figure 6).

A similar analysis can be made for European gas and, perhaps still more significantly, for oil outside the Middle East. The graph of annual world oil discoveries outside the Middle East shows a significantly declining discovery rate since about 1960 (Figure 7). Applying the 30 years lag of US oil suggests that non-Middle East oil production could well peak in the reasonably near future.

Considerable caution is needed, however, in extending use of the Huppert curve analysis from the USA to broader geographical regions. The world is quite simply a much vaster, more varied, and more variably explored area than the USA. There are many regions of the world where modern methods have barely been applied and those methods themselves are continually breaching new frontiers, such as the deep waters of the Atlantic off Brazil and Africa. Non-Middle East oil has actually been taking market share from the Middle East in recent years, where more capacity has been put on standby (some of which was used earlier in this year to help out in the tight market).

However, the declining discovery rates are troubling and indicate ample scope for oil and gas price "shocks". Moreover, in the case of oil, the impact of future shocks is likely to be more severe as oil use is now more concentrated than before in areas (such as transportation) where there is little scope for rapid substitution.

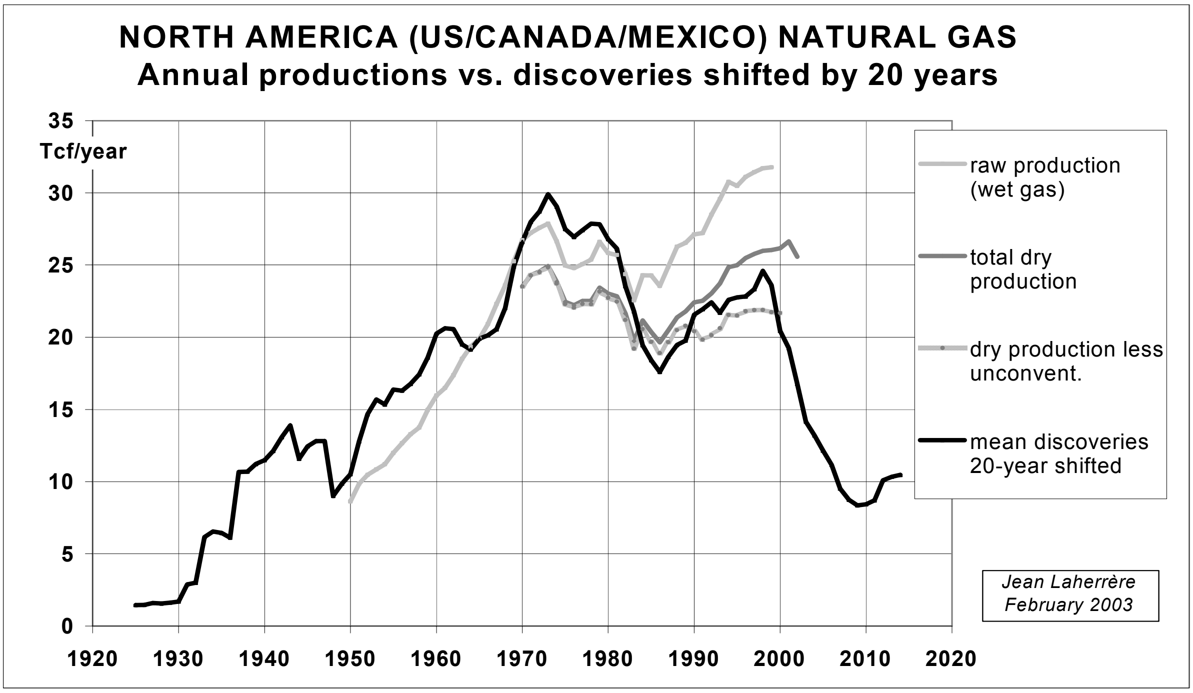

FIG. 6. Annual production vs. North American gas discoveries.

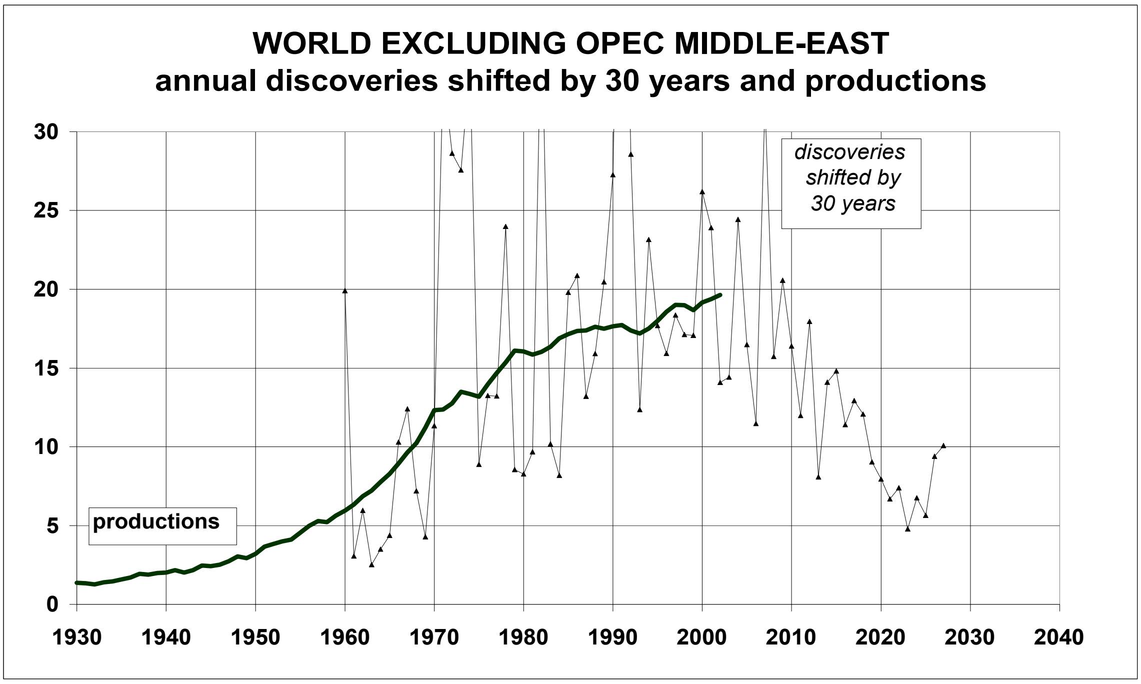

FIG. 7. Annual world oil discoveries outside the Middle East.

So the outlook would appear to be for more volatile energy markets, with price spikes for primary energies representing not so much any absolute resource scarcity as underlying shifts in energy supply profiles. Arguably the spikes will also be stronger and last longer than in the past.

What are the implications of this for different energy sources and in particular electricity? WEC has analysed energy use by the categorisation of stationary fossil fuel uses, fossil fuelled mobilty and electricity. This corresponds more closely to the notion of energy services than more the traditional categorisation of industrial, commercial, residential and agricultural. Categorised this way one can then look at the effects of primary energy price rises on their use. The effect of primary energy price rises on the price of final delivered energy is inversely related to the proportion in the price of fixed capital cost and "political charges" (taxes, environmental levies, etc). Primary energy price rises affect the final cost of stationary fossil fuel uses proportionately more than fossil fuelled mobility and electricity, where fixed and political costs are much higher. This probably explains why stationary fossil fuel uses have taken the brunt of the reduction in energy intensity seen since the first oil shock. Electricity and fossil fuelled mobility have retained a very steady relationship to GDP and have thus increased their market share (Figure 8).

If we add to this the likelihood that environmental charges will rise as – in one way or another – the externalities of fossil fuels become incorporated into their prices, the fundamentals for nuclear power of higher energy prices with a growing share for electricity look promising.

Is it enough then to wait for market fundamentals to bring about a resumption of the earlier strong growth in nuclear power? I am afraid not!

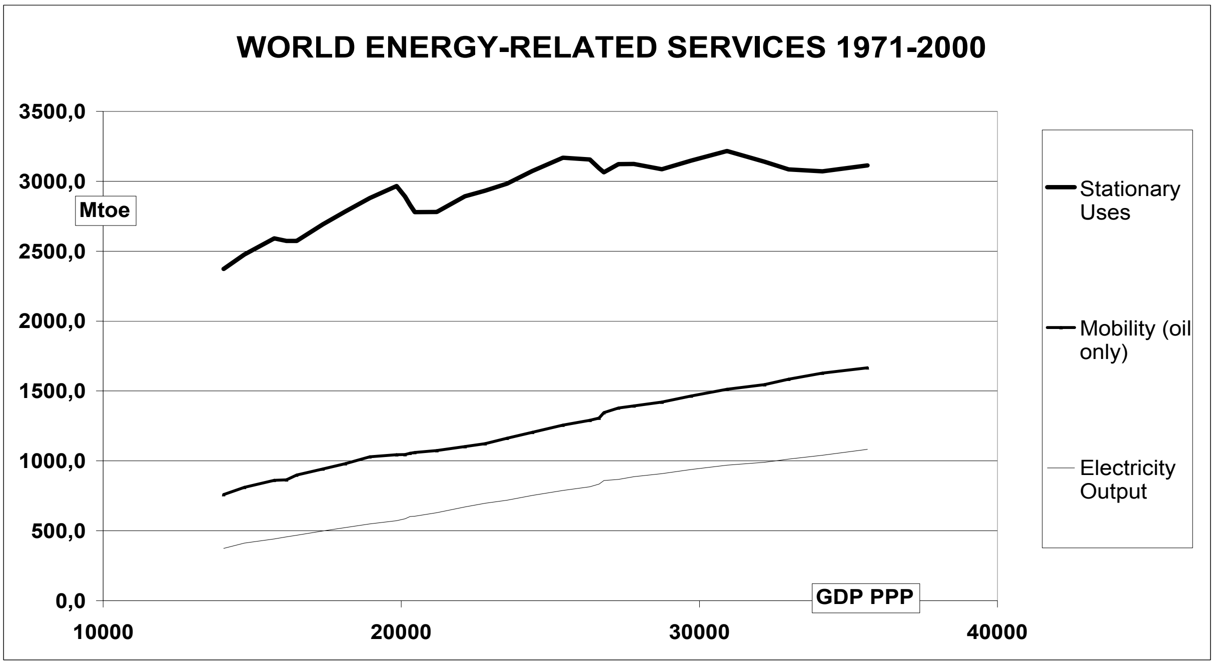

FIG. 8. Electricity and fossil fuelled mobility vs. to GDP.

While there is much re-evaluation taking place of earlier experience of electricity market reform, we do not foresee a return to the old days of central planning. We now understand more clearly that different approaches are needed at different stages of market development, but in most markets where nuclear power is applicable, it will have to compete in more or less liberalised markets and its competitors are not standing still:

— Much effort and, more significantly, money is now going into the development of near-zero and zero emissions fossil fuel systems. Projects such as President Bush's FutureGen Initiative provide the funds to test many ideas now on the drawing board. Many questions remain and, whatever happens, fossil fuels resources remain finite, but coal gasification and carbon capture and sequestration would open up the prospect of clean fossil fuel use far into the future.
— Some renewable energies are now growing fast. Certainly this is from a very low base and their total contribution to primary world energy supplies, if hydropower is excluded, is still less than 2%. However, the experience base is growing and key technologies (wind and solar in particular) are moving down the cost curve.
— The role of decentralised systems is expected to expand, partly on the back of the renewable energies and partly through fuel cells (which may or may not be renewable depending on the hydrogen source). Again there are still many questions to resolve and I, for one, do not believe decentralised power will replace the grid. However, it does seems likely that it will take increased market share.

Thus there is no room for complacency with respect to nuclear power and I stress again the importance of innovative technology in nuclear fuel cycles and nuclear power. In particular, I believe innovation needs to deliver the following:

— *Diverse product offerings to the market.* It is not enough any longer just to offer product at the maxi end of the electricity generating product range. In particular, nuclear power must be capable of meeting at least some categories of need in developing countries, or it will be a mere sideline in the energy story of the twenty first century

— *Reduced cost* The possibility of a somewhat higher energy costs should not be allowed to divert the industry from the drive to lower costs. The surest way to succeed in a competitive market is to be low cost, provided of course that this does not compromise safety.
— *Simpler and enhanced safety*. It follows from the need to respond to broader variety of electricity demand situations that simpler, more robust safety systems would be extremely helpful. For some years now, there has been an effort to start again from fundamentals, but with the benefit of the many years experience of reactor operation we have now accumulated, and to design reactors that are simpler, safer and low cost.
— *Secure waste management*. I am personally quite satisfied that waste management technologies exist which can provide safe long-term containment of high level waste, but many members of the public and also many politicians are far from convinced. Innovation to enhance the safety of planned approaches, but also innovation in communicating the safety case of these approaches, is needed.
— *Enhanced proliferation resistance*. As the concerns raised by North Korea and Iran suggest, we still need to progress further in building technical and institutional barriers to the possible use of peaceful nuclear programmes for military purposes.

From looking at the Conference programme, I see that each of these areas will be treated. I very much look forward to hearing how innovation can help ensure that nuclear energy is a substantial contributor to a carbon-constrained, sustainable energy future.

NUCLEAR ENERGY AND SUSTAINABLE DEVELOPMENT

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Abstract. In this paper the question is addressed under what condition nuclear energy may qualify as a viable option to fulfill the need for energy services of present and future generation in a sustainable manner. In this context attention is given to the following concerns: (1) public acceptance of nuclear fuel cycles; (2) safety risks of nuclear power plants and other components of the nuclear fuel cycle; (3) lifetime and management of nuclear waste, especially High Level Waste; (4) proliferation of fissile materials and nuclear weapons; (5) accumulation of radionuclides in the biosphere up to unacceptable high levels; (6) scarcity of nuclear resources; (7) cost of nuclear energy. It is concluded that these concerns must be addressed in a sustainable manner such that nuclear energy can compete on an economic basis. In this paper approaches, concepts and ideas are formulated that may help to fulfill this criterion.

1. PREFACE

The organizers of the 'International Conference on Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power' (Vienna, Austria, 23-26 June 2003), have invited me to present my views on nuclear energy and sustainable development in the context of the INPRO-study of IAEA [1], based on an essay I wrote in 1996 – as part of a technology assessment of the HTR – to discuss requirements that nuclear energy technologies and fuel cycles should fulfill to be able to contribute to a sustainable development of our society [2].

In this paper I won't discuss (1) the issue of sustainable development as such, (2) the different dimensions of sustainable development, (3) the relation between sustainable development and energy demand and supply, and (4) the need for a new energy strategy to achieve sustainability. These discussions can be found in my 1996-essay, but also in e.g. the World Energy Assessment, published in 2000 [3], in which I participated editorial board member and one of the convening lead authors. Ideas can also be found in early reports of the UN Committee on New and Renewable Sources of Energy and on Energy for Development (UN-CNRSEED) [4, 5] and the reports of the UN Committee on Energy and Natural Resources for Development (UN-CENRD) [6, 7] in which I participated as chairman of the Subgroup on Energy till December 2002.

Instead, in this paper, I will focus directly on the issue of nuclear energy and sustainable development: under what conditions may nuclear energy qualify as a viable option to fulfill the need for energy services of present and future generation in a sustainable manner.

2. INTRODUCTION

One of the major issues to be solved to achieve a sustainable development of economic, social and ecological systems is the threat of a severe climate change within a short period caused by human behavior. The emission of greenhouse gases, especially CO_2, due to the manner in which we fulfill our demand for energy services, is the major cause.

Nuclear energy provides about 16% of the amount of electricity produced globally. It contributes about 7% of the global primary energy demand (29 EJ of 418 EJ in total in 2001). In the year 2001 about 440 nuclear power plants, operating in thirty-one different countries, had a total installed capacity of nearly 360 GWe and generated about 2500 TWh electricity [1]. Almost four-fifths of these power plants are of the Light-Water type.

The short-term prospect for nuclear power is opaque. There is no strong increase in installed nuclear capacity at present. Although the annual additions in installed capacity arrived at peak values of about 30,000 MWe in the mid eighties, this figure decreased to some thousand MWe in the nineties. In fact, new nuclear power plants are not built fast enough to maintain nuclear power's 16% share of global electricity generation [1], as the annual increase of the nuclear output with about two and a half percent a year is less than the three to four percent growth rate of the world's electricity consumption. Most analysts project that nuclear energy's contribution to the global energy budget will not grow and may even decline during the initial decades of the 21st century.

Without nuclear energy, the present global emissions of CO_2 due to our energy consumption would have been about 7% higher, depending on the fossil fuel use that is replaced. For nuclear energy to make a significant contribution to coping with climate change, nuclear capacity must be increased by at least an order of magnitude. Technically, this is feasible, as nuclear energy could contribute 200 EJ or more per year to the world energy consumption in the second half of the 21st century [3, 7, 8].

Further application of nuclear energy could contribute not only to the reduction of greenhouse gas emissions, but also to a decreased emission of other pollutants like sulfur dioxide, nitrous oxide, small particulates and volatile organic compounds. It would also help to limit the dependence on fossil fuels, to meet security of supply concerns, and to guard against potentially escalating fossil fuel prices.

Opposite these advantages, there are a number of disadvantages. For many people these disadvantages are too big to make nuclear energy an attractive option. Consequently, in their view nuclear energy should not receive much attention when striving for sustainable development.

From debates on the pros and cons of using nuclear energy, it can be concluded that the resistance and (perceived) disadvantages are related mainly to the following issues:

1. Public acceptance of nuclear fuel cycles;
2. Safety risks of nuclear power plants and other components of the nuclear fuel cycle;
3. Lifetime and management of nuclear waste, especially High Level Waste;
4. Proliferation of fissile materials and nuclear weapons;
5. Accumulation of radionuclides in the biosphere up to unacceptable high levels;
6. Scarcity of nuclear resources;
7. Cost of nuclear energy;
8. Industrial development (local capacities, customers interest, spin offs, employment);

9. Lock-in effects (impact on development of non-nuclear options).

In this paper I will focus on the items 1-7.

3. PUBLIC ACCEPTANCE OF NUCLEAR FUEL CYCLES

In many countries public acceptance of using nuclear power to fulfill our energy needs is small. In the Netherlands – for example – the first public resistance against building new nuclear plants came up around 1970. In the mid seventies, opinion polls indicated that the resistance was increased such that about 50% of the population of the Netherlands was against building new nuclear plants. In 1979 this percentage increased to 85% by the accident with the nuclear plant in Harrisburg. This level was maintained till 1984, withstanding the high costs of oil and natural gas as well as the Social Debate on Nuclear Energy in the period 1981 till 1983 initiated by the Dutch Government. Despite this opinion, the government decided in 1984 to build three nuclear power plants. Soon thereafter the percentage of the population against new plants dropped to the level before the Harrisburg accident occurred (about 50%), but increased again in 1986 due to the nuclear disaster in Tsjernobyl to 85% [9]. Since then the resistance remained on this level. Only very recently there are indications that it may have decreased somewhat, as in 2002 a resistance of about 80% was found. However, the 2002 investigation also indicates that not more than 4% of the population selects nuclear energy as the favorable technology to generate electricity [10].

The size and persistence of the resistance leads to the question whether it will ever be possible to obtain public acceptance of nuclear power again. A number of years ago this was investigated by SMO, the Netherlands Foundation on Society and Business [11]. In this research, called “Nuclear Energy and the Environment”, 404 persons selected randomly were first informed about the state of the art and future potential of nuclear energy technology, thereafter a survey was made about there opinion. From this survey it appears that about $^2/_3$ of the people would accept the application of nuclear energy to generate electricity if the technology would be much saver, preferably inherently safe. However, only $^1/_3$ of the people believe that (inherently) safe nuclear power plants can be constructed. Another result of the survey is, that there is a lack of trust in nuclear experts. Only ¼ of the people believe that these experts will inform the public enough about potential dangers of (new) nuclear energy technologies. The results show that public acceptance of nuclear energy may come back if the technology becomes safer, preferably inherently safe. However, one doesn’t believe that this is possible, nor does one trust nuclear experts if they tell that (concepts of) new nuclear plants can fulfill this safety requirement.

The survey also indicates that in energy policy high priority should be given to the potential of energy efficiency improvement and to renewables. More recent research also indicates that preference is given to CO_2 capture and storage technologies above nuclear technologies when there is a need to reduce CO_2 emissions [26].

From the survey and from other analyses in the field it can be concluded that public acceptance of nuclear technology to generate electricity or other energy carriers may come back only when a number of conditions are satisfied [2]:

- Full consideration of the potential and use of energy efficiency improvement measures, renewable energy technologies and advanced fossil fuel technologies. Activities in the field of nuclear energy development shouldn’t block these developments.
- Development of nuclear technologies that are inherently safe, clean, more proliferation resistant, economic, and efficient when dealing with scarce resources.

— Regain of trust in nuclear experts as well as nuclear decision-making processes. It requires amongst others the creation of ample opportunity for public participation in decision-making processes on nuclear energy. It also requires that public opinions about nuclear energy should be taken serious. Also it is important that the role of governments and agencies dealing with the control of nuclear technologies is not merged with activities to promote the use of nuclear energy.

4. SAFETY RISKS OF NUCLEAR POWER PLANTS AND OTHER COMPONENTS OF THE NUCLEAR FUEL CYCLE

Public concern about nuclear safety arises from the potential for sever accidents involving the release of substantial quantities of radio-nuclides produced in nuclear reactors and causing damage to people, communities, economies and the environment. Striving for sustainability means for the whole nuclear fuel cycle, from cradle to grave, that on a regional or higher level *no* exhaustion, affection, pollution or use of space should take place such that the fulfillment of primary needs like nutrition, clothing, sheltering and mobility is endangered for present or future generations [12]. In policy making concerning external safety, the concept *no* is translated to a probability for severe damage; this probability should be small enough, in principle *negligible* small, and certainly not *unacceptable* high. In the Netherlands this approach is translated into norms and standards to protect (1) each person staying in the neighborhood of a facility that may cause danger, and (2) to limit he risk of social disruption due to a severe accident with this facility. This is translated in two standards: (1) the 'individual risk' (nowadays called 'location connected risk') and (2) the 'groups risk' (which regulates the probability of accidents with at least 10 acute fatalities).

The General Energy Council of the Netherlands advised that, "if new nuclear plants are built, it should be required that the probability of a severe accident with these plants should be negligible small, for individuals as well as groups of people" [13]. For individual persons this criterion implies that the probability to die due to a severe accident should be less than 10^{-8} per year. For groups of people this criteria means that the probability of an accident with 10 deaths or more should be less than 10^{-7} per year, with 100 deaths or more less than 10^{-9} per year, with 1000 deaths or more less than 10^{-11} per year, etcetera.

However, a probabilistic approach to risk assessment has some problems and draw backs:

— The acceptability of very severe accidents with large amounts of victims and severe social disruption for a long period of time is not excluded in principle.
— The accuracy of calculating safety risks is small and the uncertainty high, especially in case of very small probabilities and very high effects.
— Based on technologies used today, the risk of a severe accident is strongly influenced by the possibility that people don't use these technologies in a proper way. Consequently, the validity of risk calculations can get lost if the social context within which the technology has to function is changed drastically.
— To reduce risks, a well-known approach in policy making in the ALARA-principle. This principle implies that permanently we should opt for the development and application of the safest technology, if reasonably achievable.

Taking into account the social debate on nuclear energy and the objections against the risk approach formulated above, one might wonder whether the approach suggested by the

General Energy Council of the Netherlands will result in enough public support to build nuclear power plants. In addition one can doubt that the risk of very severe accidents with many victims and extensive social disruption is acceptable at all when striving for a sustainable future, especially if the availability of other options to fulfill our energy demands reduces the need to take this risk.

Consequently, a risk approach is required which is based much more on deterministic instead of probabilistic arguments to assure safety. A central characteristic of this approach should be that concepts and technologies are applied being *inherently safe*. This requirement should apply for each part of the nuclear fuel cycle, not only the power plant but also the reprocessing plant and facilities to store High Level Waste and fissile materials.

With the term 'inherent safety' we mean that it can be demonstrated based on physical principles, that installations containing large amounts of fissile materials and nuclear waste *cannot melt or explode*, whatever may happen. Note that inherently safe doesn't mean that nothing can go wrong.

In the case of nuclear power plants, the requirement of inherent safety means that under no condition fissile material or fission products can escape from the rods after loss-of-coolant or a change in reactivity. That implies setting conditions that should be fulfilled concerning amongst other:

— Energy production per amount of fissile material (J/kg), that should not be too large;
— Power density of the reactor core (W/m^3), that should be small enough;
— Heat capacity of the core of the reactor (J/K), that should be large enough;
— Surface/volume ratio of the core (m^2/m^3), that should be large enough;
— Maximum allowable increase of reactivity ($/sec);
— Reactivity coefficient, power coefficient and void coefficient should all be negative;
— Chemical and physical properties of the materials used in the reactor.

Further analysis should indicate what the requirements exactly should be. As an example, the requirement for the power density might be that this density should be less than about 5 MW/m^3. An HTR (about 3 MW/m^3) can fulfill such a condition, but not a PWR (about 100 MW/m^3), nor an ABWR (about 50 MW/m^3) or a SBWR (about 40 MW/m^3).

The HTR can be constructed such that a loss-of-coolant accident won't result in a melt down. This makes this reactor type interesting from a safety point of view, provided that it can be demonstrated that this type of reactor can also not explode, whatever may happen, and that the materials used in the core of the HTR – like graphite – never compromise the inherent safety requirement. In interesting concept of this reactor type is the 100 MW Pebble Bed Modular Reactor that ESKOM (South Africa) is developing.

However, other concepts and designs of nuclear reactors - like the accelerator type - may even be more attractive from a safety point of view [14].

In addition it should be demonstrated that also other parts of the fuel cycle, like reprocessing and storage, can be operated inherently safe too.

In the above, the focus is on accidents caused internally. However, the nuclear fuel cycle should also be safe for attacks coming from the outside, like terrorist groups.

5. LIFETIME AND MANAGEMENT OF NUCLEAR WASTE, ESPECIALLY HIGH LEVEL WASTE

Nuclear waste management and disposal is probably the issue where the gap between nuclear supporters and opponents is widest. Many in the industry believe that technical solutions are available but are being blocked politically, while others see this impasse as a reason to block nuclear power use [7]. Most concern focuses on the disposal of high-level nuclear waste (aaaspent fuel to meet the safety requirements applicable in different counties, although there are technical uncertainties that need further study [3,7]. However, many others have the opinion that nuclear power is incompatible with sustainable development because its HLW may remain hazardous over tens to hundreds of thousands of years. It indicates that the nuclear waste issue is probably not primarily a physical but a moral issue. Further investigation and development of approaches and technologies to reduce the lifetime of HLW to hundreds of years might help to solve this issue. To achieve that, the following options may play a role:

— Full partitioning of uranium, plutonium and actinides from spent fuel and their use in nuclear reactors;
— Transmutation of long-lived radionuclides into short-lived ones, using MeV neutrons from nuclear reactors or GeV protons using accelerators, possibly combined with energy supply technologies;
— Use of thorium instead of uranium as feedstock for nuclear fission processes.

At present, however, the future of waste transmutation doesn't look bright. A European Parliament's report notes that it has obvious attractions, but it is perhaps "just nice physics" rather than an economically and technically practical option. It would also be a slow process. Moreover, only some waste would be suited to transmutation [15]. However, another study concludes: "Although some of the current concepts proposed are far from being realized, considerable scope seems to exist for opportunities to reduce the presently produced amounts of nuclear waste. These ought therefore to be further investigated" [16].

6. PROLIFERATION OF FISSILE MATERIALS AND NUCLEAR WEAPONS

A 1000 MWe Light Water Reactor discharges about 200 kg of plutonium a year. This amount is enough to produce twenty nuclear explosives. A global installed LWR capacity of 3,000 GWe would produce about 500.000 kg of plutonium a year. With breeder reactors instead of LWR's, this figure could increase to about 5 million kg plutonium a year. This would create a high diversion risk of fissile materials and nuclear weapons [3]. In addition, also in others parts of the nuclear fuel cycle technologies and materials can be misused, i.e. applied for non-peaceful purposes. Therefore, a nuclear system should be developed that is far more diversion resistant. It requires stronger institutional arrangements to keep peaceful and military uses separate and to prevent the misuse of nuclear possibilities. It also requires the development and application of advanced technologies aimed at limiting opportunities of acquiring nuclear weapons under the guise of peaceful nuclear energy applications and stealing weapon-usable nuclear materials.

In this context, Williams and Feiveson [17] formulated a number of "diversion-resistance criteria for future nuclear power" that - somewhat modified - can be summarized as follows:

(i) Development of an advanced nuclear reactor and fuel cycle technology that produces far less fissionable, weapons-usable materials in spent fuel; as an indication, less than a critical mass per year per GW of capacity.

(ii) The new technology should in principle be applicable in each modern society in a sound manner, 'culture proof' and without discriminatory conditions among nations.

(iii) Fissionable weapons-usable material that is not contained in spent fuel and facilities to enrich uranium or to separate plutonium shall not exist outside international centers that are maintained under tight physical security of the IAEA.

(iv) As far as possible, fissionable weapons-usable material produced in reactors should be contained in spent fuel.

(v) Spent fuel shall be stored in international centers (see above).

There has been some debate on these proposals in the literature, but not enough to come to final conclusions and a plan of action. Implementation of these criteria would not result in a definite prevention of the misuse of fissile materials and proliferation of nuclear weapons due to peaceful use of nuclear energy. No technology or organizational structure can achieve that. However, if these approaches were followed one may expect that the peaceful use of nuclear energy is an un-attractive route to acquire nuclear weapons.

Applying the HTGR technology and using thorium would reduce the production of plutonium considerably. Per GWe one should think about a discharge in the spent fuel of about 30 kg a year, versus about 200 kg a year in the case of conventional LWR's. That would be an important improvement, but it's not enough according to the criteria given before; the production should be well below 20 kg of plutonium a year [17].

In the literature proposals can be found for nuclear fission plant designs that would hardly contribute to the proliferation of nuclear weapons. Often these designs are based on the use of proton accelerators. An example is the Linear Accelerator Breeder concept, proposed by Steinberg in 1977, which would "decouple fuel production from weapon-type operations of enrichment and reprocessing, hold out the possibility of environmentally congenial disposal of radioactive waste, and provide assurance of an unlimited supply of safe and sound energy" [18]. More recently, a variant of this concept was proposed by Rubia, called the Energy Amplifier. It is unclear to what extent these concepts are viable, see also [14, 19]. However, the suggested advantages require full understanding of the potential, merits and drawbacks of the linear accelerator approach. In a status report of IAEA on accelerator driven systems (ADS), published in 1997, it is concluded that ADS offer a sub-critical mode of operation for nuclear power systems, reduce long-lived waste burden, may ensure a proliferation-resistant fuel cycle and treatment, open a promising way to utilize thorium resources, and offer an alternative way to utilize the existing excess of weapons-grade plutonium for energy production. Because of environmental reasons "ADS is an attractive option which research institutes and industry should develop and offer to the international community" [20].

7. ACCUMULATION OF RADIONUCLIDES IN THE BIOSPHERE UP TO UNACCEPTABLE HIGH LEVELS

The use of nuclear energy is associated with the emission of radionuclides into the biosphere. These emissions may result in an accumulation of these nuclides in time and in parts of the biosphere, depending on physical, chemical and biological properties of these nuclides. Due to accumulation, the emissions may cause health damage on the longer term and influence the functioning of natural systems negatively. Therefore, this aspect should be considered carefully when developing a nuclear system with a globally installed capacity of 3,500 GWe or more. Such a development should not have irreversible negative effects for present and future generations.

One of the radionuclides deserving specific attention is Krypton-85, a gaseous fission product that is emitted during the reprocessing of spent fuel. It accumulates in the atmosphere. This accumulation can cause changes in the electric and chemical properties of the air, influencing atmospheric processes like lightning and the creation of clouds. Consequently, on the long term the emission of Kr-85 may influence the weather and climate on earth [21]. To prevent that, the emission of Kr-85 should be limited, depending on the growth of nuclear capacities. It is up to the nuclear industry to indicate how, and on what costs, deep reductions of Kr-85 emissions can be achieved.

Attention should also be given to the accumulation of other radionuclides that may cause damage. Examples are tritium (H-3), Jodium-129 (J-129) and Carbon-14 (C-14). The production and discharge of C-14 may cause problems especially if in nuclear reactors lots of Carbon were used, e.g. as moderator in HTR's.

8. SCARCITY OF NUCLEAR RESOURCES

An efficient use of scarce resources is one of the characteristics of sustainable development. In addition, the need to reduce the production of waste implies that also materials should be used very efficiently. However, this should not compromise requirements formulated with respect to safety, nuclear waste management and proliferation.

Within this context, it should be demonstrated that we have enough resources to apply nuclear energy in a meaningful way. Assuming a globally installed nuclear capacity of 3,500 GWe, it should be possible to use this capacity for at least a number of generations (consequently, well beyond the year 2100).

From a study we did in 1989 on "Nuclear Energy and the CO_2 Problem" it was concluded that the ultimately recoverable resources of uranium might be estimated at about 30 million ton, whereas the known recoverable resources were about a factor 10 less [22]. In essence, these figures still apply. The known recoverable resources of uranium are estimated at present at about 3.3 million ton, the ultimately recoverable resources at 30 – 100 million ton, depending on the acceptance of a steep increase of the uranium price [3, 15, 23].

If the generation of 1 MWh_e requires 24 gram of uranium – which is the case in a LWR assuming a once through approach - 30 million ton would allow the generation of 125 x 10^4 TWh_e, about 100 times the present annual world electricity production. If, as a result of reprocessing, the fuel cycle would require 12 gram of uranium per MWh_e, this figure would be 200 times the present annual electricity production.

It should be noted that in modern power plants the efficiency of fissile material use will probably be better than assumed here. It certainly increases strongly if breeder concepts are applied.

Also it should be noted that the nuclear fuel cycle consumes fossil fuels, from cradle to grave, causing indirect emissions of greenhouse gases. However, these emissions are far less than the greenhouse gas emissions of conventional natural gas or coal plants. From scenario studies it is concluded that the greenhouse gas mitigation potential of nuclear energy could be reductions in CO_2 emissions of 100 - 300 GtC during the next hundred year – reductions that may be equivalent to about 10 – 20% of the emissions under a business-as-usual future [3, 8, 22].

Finally, it should be noted that beside uranium also thorium could be used to generate electricity. The recoverability of thorium is at least as good as uranium, probably even better [3, 15].

Consequently, it is concluded that scarcity of resources it not an argument to disregard the nuclear option as a potential source of energy to be applied to achieve sustainable development.

9. COST OF NUCLEAR ENERGY

Nuclear energy has the advantage that large amounts of energy can be released from small amounts of relatively abundant and cheap material. Consequently, in the period after the Second World War, there were suggestions that nuclear electricity would be 'too cheap to meter' [14].

The reality has proved somewhat different. Fifty years on, after very large investments, the cost of the electricity produced remains high. Only under special conditions, nuclear energy has proved to be able to compete well with fossil fuelled alternatives.

At present the economics of nuclear power is likely to become even less favorable, as for the alternatives costs have coming down if competitive conditions are strong. Moreover, in the current context of liberalization of the electricity and energy market, nuclear's capital intensity constitutes a clear disadvantage. Consequently, at current and expected gas prices, new nuclear power plants cannot compete against natural gas-fuelled combined cycle technologies in those places where gas supply infrastructures are in place [3, 16, 24].

Over the years, new nuclear power plants have become progressively more capital intensive, taken longer to build than other conventional power generating facilities, involved increasingly prescriptive and cumbersome procurement, and entailed longer and costlier regulatory and licensing procedures. All these factors tend to increase financial and commercial risks, and delay innovation [25]. On average, the capital costs for building new nuclear plants of current reactor design are 2-4 times more than fossil-fuelled plants. The challenge for the industry is to reduce these costs to a generally competitive level. Without innovation, nuclear power is unlikely to meet this challenge [7].

Quantification of the external costs of today's fossil energy plants would improve the economics of nuclear plants. But these benefits will not be so great with various advanced fossil fuel technologies involving fuel decarbonization and CO_2 sequestration [3, 26]. Thus direct economic costs will continue to be important in determining the future of nuclear power.

In conclusion, as stated in the World Energy Assessment published in 2000: "If nuclear power is to become economically viable again, innovations will be needed that can provide electricity at costs competitive with other future near-zero-emission energy technologies. Moreover this has to be done in ways that are consistent with meeting concerns about nuclear safety, proliferation and diversion, and radioactive waste disposal" [3].

10. EPILOGUE

In summary, based on the approaches and conditions formulated in this paper, it is concluded that present day nuclear technology use is not compatible with sustainable development. For nuclear energy to qualify as a sustainable energy option, concerns regarding safety, waste management and disposal, proliferation and diversion, and public acceptance must be addressed in ways that enable nuclear energy to compete on an economic basis. It requires new concepts and ideas, technological innovation, as well as improved institutional arrangements and risk management strategies. Advanced technology development that addresses effectively the concerns expressed in this paper is also a prerequisite to enable large expansion of the installed nuclear capacity.

In this contribution I have tried to formulate approaches, concepts and ideas that may help nuclear energy supply to qualify as a sustainable energy option. Some of them are addressed in the INPRO study, some not. The proposals made in the INPRO study are important, but only a first step toward sustainability. Further steps are needed to address the issue of sustainability as a leading principle for nuclear energy developments fully.

11. REFERENCES

[1] INTERNATIONAL PROJECT ON INNOVATIVE NUCLEAR REATORS AND FUEL CYCLES (INPRO): "Guidelines for the Evaluation of Innovative Nuclear Reactors and Fuel Cycles (Final Draft)", Phase-1A Report – Extended Version, IAEA, Vienna, Austria, May 2003.

[2] TURKENBURG, W.C., "Nuclear Energy and Sustainable Development", Report ECN-C-96-048, ECN, Petten, The Netherlands (1996) (in Dutch).

[3] WORLD ENERGY ASSESSMENT (WEA): "Energy and the Challenge of Sustainability", UNDP/UN-DESA/WEC, New York (2000).

[4] UNITED NATIONS COMMITTEE ON NEW AND RENEWABLE SOURCES OF ENERGY AND ON ENERGY FO DEVELOPMENT (UN-CNRSEED): "Report on the First Session of the Committee on New and Renewable Sources of Energy and on Energy for Development", ECOSOC, United Nations, New York, E/1994/25 (1994).

[5] UNITED NATIONS COMMITTEE ON NEW AND RENEWABLE SOURCES OF ENERGY AND ON ENERGY FO DEVELOPMENT (UN-CNRSEED): "Report on the Second Session of the Committee on New and Renewable Sources of Energy and on Energy for Development", ECOSOC, United Nations, New York, E/1996/24 (1996).

[6] UNITED NATIONS COMMITTEE ON ENERGY AND NATURAL RESOURCES FOR DEVELOPMENT (UN-CENRD): "Report on the Second Session of the UN Committee on Energy and Natural Resources for Development" - ECOSOC, Official Records, United Nations, New York (1999).

[7] UNITED NATIONS COMMITTEE ON ENERGY AND NATURAL RESOURCES FOR DEVELOPMENT (UN-CENRD): "Report on the Second Session of the UN

Committee on Energy and Natural Resources for Development" - ECOSOC, Official Records, United Nations, New York (2000).
[8] INTERGOVERNMENTAL PANEL ON CLIMATE CHANGE (IPCC): "Special Report on Emission Scenarios", Cambridge University Press, Cambridge, UK (2000).
[9] TURKENBURG, W.C., "Nuclear Energy, refused and embraced" Seminar on 'Energy and Society', Directorate General on Energy, The Hague, The Netherlands (1990) (in Dutch).
[10] NIPO, cited by: NRC-Handelsblad, 17 May 2003, page 17 (in Dutch).
[11] SMO (NETHERLANDS FOUNDATION ON SOCIETY AND BUSINESS): "Results Third Investigation on Technology and Society: Nuclear Energy and the Environment", Stichting Maatschappij en Onderneming, Den Haag, The Netherlands (1990) (in Dutch).
[12] JANSEN, J.L.A., "Sustainability and the Safety of Nuclear Energy", 'Proceedings of a Workshop on Nuclear Energy and Sustainable Development', Dept. of STS, Utrecht University, Utrecht, The Netherlands (1994) (in Dutch).
[13] AER (GENERAL ENERGY COUNCIL OF THE NETHERLANDS): "Energy and Sustainable Development", Algemene Energie Raad, The Hague, The Netherlands, 1989 (in Dutch).
[14] ANDRIESSE, C.D., "Sustainability and Radioactive Waste", 'Proceedings of a Workshop on Nuclear Energy and Sustainable Development', Dept. of STS, Utrecht University, Utrecht, The Netherlands (1994) (in Dutch),
[15] ELLIOTT, D. "The Future of Nuclear Power", in: 'Energy Systems and Sustainability', Open University, London, 2003.
[16] BRUGGINK, J.J.C., VAN DER ZWAAN, V.C.C., "The role of nuclear energy in establishing sustainable energy paths", Report ECN-C-01-109, ECN, Petten, The Netherlands, October 2001.
[17] WILLIAMS, R.H., and H.A. FEIVESON: "Diversion-resistance criteria for future nuclear power", Energy Policy, July/August 1990, pp. 543-549.
[18] STEINBERG, M., "Nuclear power without nuclear bombs", New Scientist, 7 July 1977, pp 14-16.
[19] WEINBERG, A.M., "Nuclear energy and proliferation: a longer perspective", in: A. Weinberg et al. (eds.), 'The Nuclear Connection: A Reassessment of Nuclear Power and Nuclear Proliferation"", Paragon House Publishers, New York, USA (1985) pp. 221-239.
[20] INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA): "Accelerator driven systems: Energy generation and transmutation of nuclear waste – Status Report", IAEA-TECDOC-985, IAEA, Vienna, Austria, November 1997.
[21] DE JAGER, D., "Effects of Radionuclides in the Atmosphere on the Weather, Climat and Environment", Publication 1992/4, Ministry of Public Housing, Spatial Planning and Environment, The Hague, The Netherlands (1992) (in Dutch).
[22] BIJLSMA, J., BLOK, K., TURKENBURG, W.C., "Nuclear Energy and the Carbon Dioxide Problem", Dept. of STS, Utrecht University, Utrecht, The Netherlands (1989) (in Dutch).
[23] ROYAL SOCIETY / ROYAL ACADEMY OF ENGINEERING: "nuclear Energy: the Future Climate" (1999), http://www.royalsoc.ac.uk/policy/reports.htm.

[24] INTERGOVERNMENTAL PANEL ON CLIMATE CHANGE (IPCC): "Third Assessment Report Workgroup III", Cambridge University Press, Cambridge, UK, 2001.

[25] ROGNER, H.H., and L.M. Langlois: "The Potential Role for Nuclear Power in a Sustainable Energy Future", Paper prepared for the UN-CENRD, IAEA, Vienna, Austria, August 2000.

[26] TURKENBURG, W.C., "Sustainable Development, Climate Change, and Carbon Dioxide Removal"- Energy Convers. Mgmt, 38 (1997) pp. S3-S12.

ECONOMIC REQUIREMENTS FOR INNOVATIVE TECHNOLOGIES – *Trends in U.S. Nuclear Power Production Costs*

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Abstract. Over the past 30 years, U.S. nuclear power production has steadily increased from 83.5 billion kilowatt hours per year in 1973, to almost 800 billion kilowatt hours in 2002. New plants accounted for a significant portion of that increase. Since the last plant came on-line in 1996, the increase in nuclear power production has come primarily from improved plant capacity factors. This has been accomplished through improved plant operating performance, reduced refueling outage durations and power up-rates. Over the last 15 years, great strides have been made in reducing and controlling plant operating costs, especially labor costs. Numerous U.S. nuclear plants have gone through downsizing efforts and hiring freezes. As a result, the U.S. has seen a steady decline in nuclear production costs on a cents per kilowatt-hour basis. While we have continued to see steady improvement in the economics of U.S. nuclear power over the past decade, our ability to sustain that level of improvement will be challenged in the future mainly due to increases in security, materials integrity issues, nuclear property insurance and employee benefit costs. Future economic survival will depend on the industry's ability to create innovative solutions for controlling costs through process improvements and un-tapping increased generation potential.

1. INTRODUCTION

Wolf Creek's 10,500-acre site is in Coffey County, four miles northeast of Burlington, Kan., 55 miles south of Topeka, Kan., 90 miles southwest of Kansas City, Mo., and 120 miles northeast of Wichita, Kan.

Wolf Creek is a Westinghouse four-loop pressurized water reactor (Figure 1) producing more than 1,200,000 kilowatts of electricity per hour. "Four loop" means that there are four steam generators plus related piping and equipment. Service areas include most of Kansas and western Missouri. Wolf Creek Nuclear Operating Corporation was organized in 1986 to operate, maintain, repair and eventually decommission Wolf Creek Generating Station.

Wolf Creek is a Standardized Nuclear Unit Power Plant System plant (Figure 2). Wolf Creek and the Callaway plant in Missouri are virtually identical. Construction of Wolf Creek began in 1977. After eight years of construction, completion and startup, Wolf Creek was declared to be in commercial operation Sept. 3, 1985. Our 2002 three year average capacity factor was just over 88 percent. Our 2002 three year average production costs were 1.48 cents per kWh. The longest continuous operating run at Wolf Creek was 487 days.

Over the past 30 years, U.S. nuclear power production has steadily increased from 83.5 billion kilowatt hours per year in 1973, to almost 800 billion kilowatt hours in 2002 (Figure 3). The new plants that came on-line in the 1980s accounted for a significant portion of that increase. Since the last plant came on-line in 1996, the increase in nuclear power production has come primarily from improved plant capacity factors. This has been accomplished through improved plant operating performance, reduced refueling outage durations and power up-rates.

U.S. capacity factors increased from 57.6 percent in 1980, to 90.7 percent in 2001 (Figure 4). Much of this improvement can be attributed to shorter refueling outages.

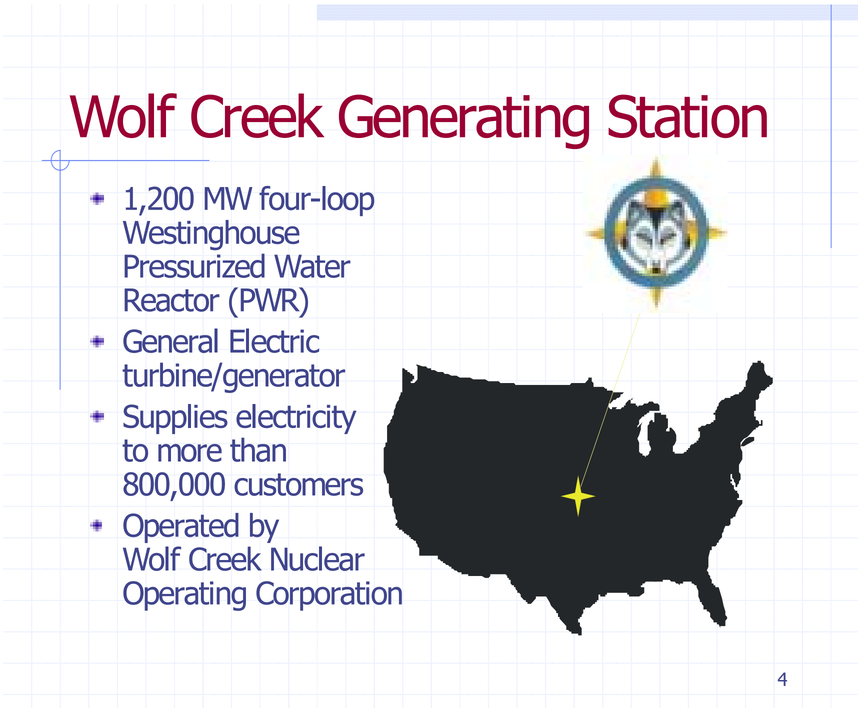

FIG. 1. Main characteristics of Wolf Creek generating station.

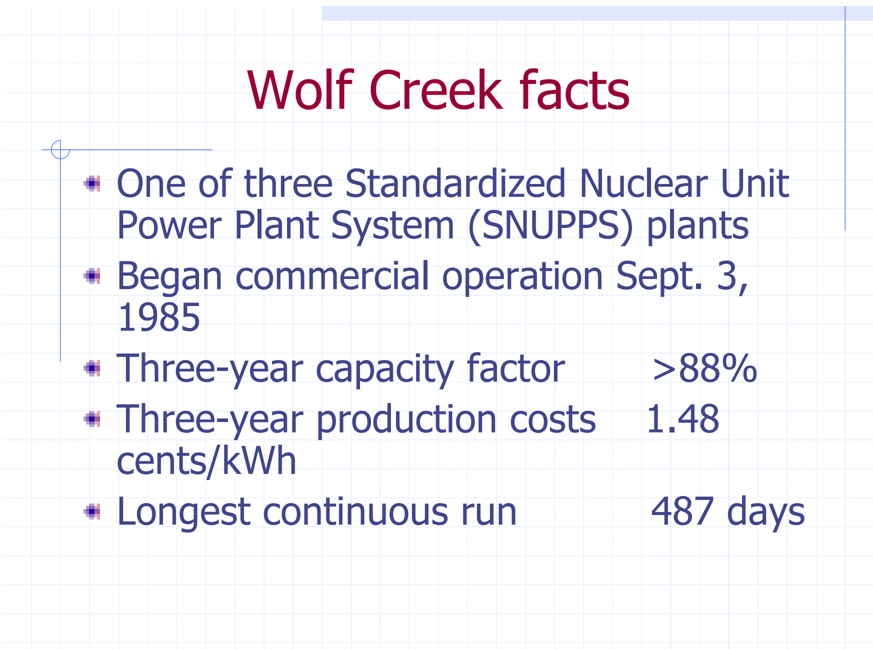

FIG. 2. Wolf Creek - standardized nuclear unit power plant system plant.

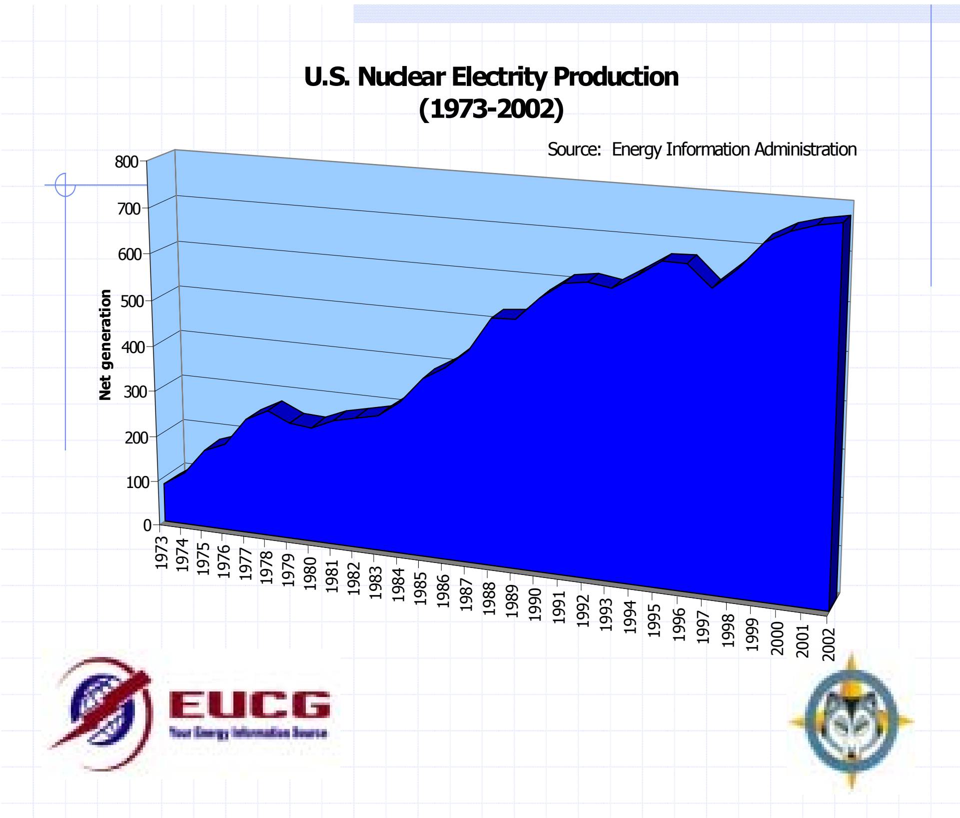

FIG. 3. US nuclear electricity production.

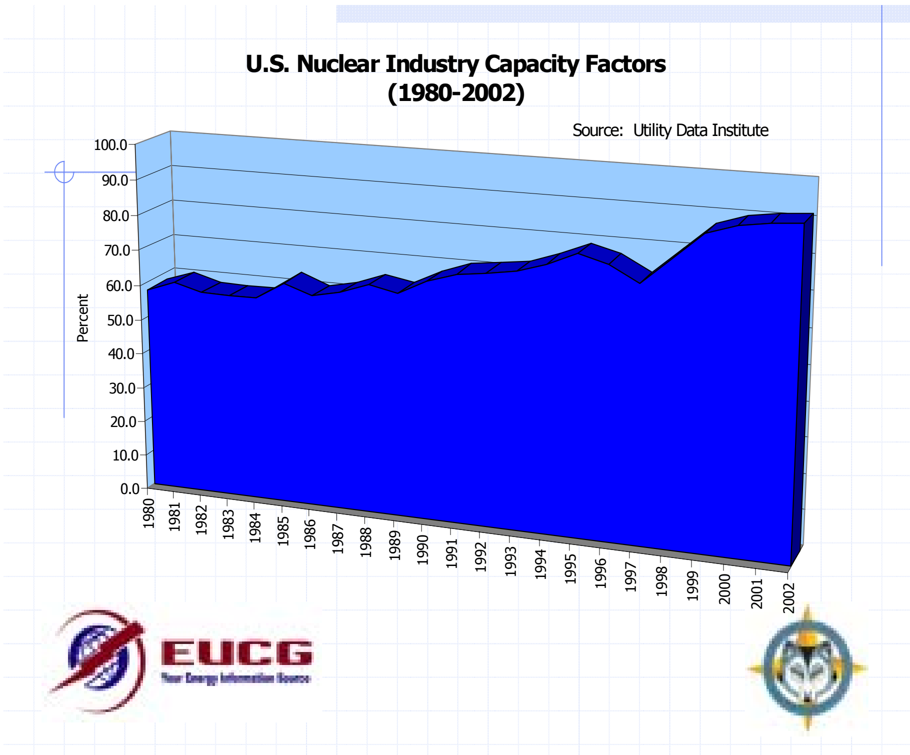

FIG. 4. US nuclear industrey capacity factors.

The average refueling duration in 1990 was 105 days (Figure 5). In 2001, refueling outage duration had been reduced to an average of 37 days. Another factor that influenced improved capacity factors was better plant operating performance.

Unplanned capability loss factors dropped from 11.6 percent in 1980 to 1.6 percent in 2001 (Figure 6). Shorter outages and fewer forced outages have both contributed to continued increase in U.S. nuclear power production.

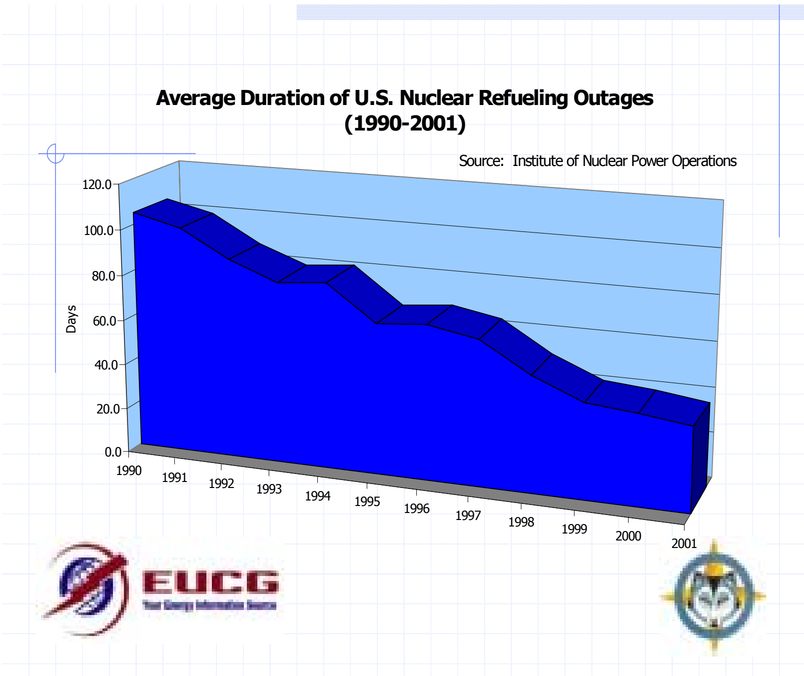

FIG. 5. Average duration of US nuclear refueling outages.

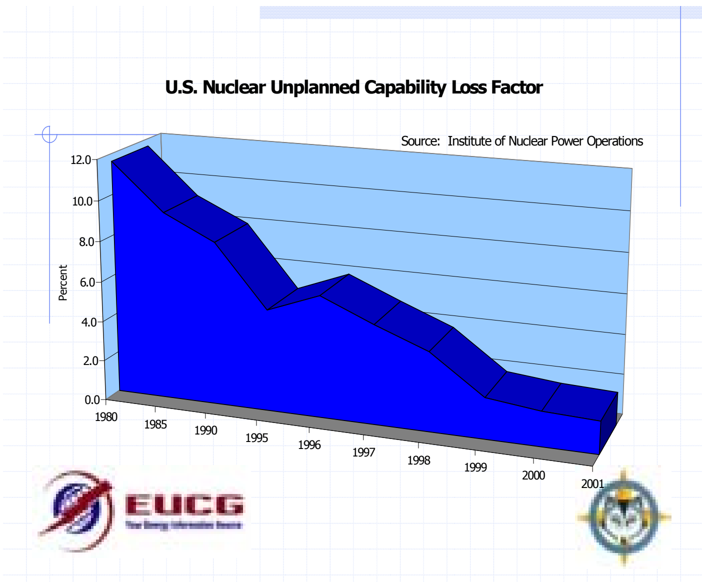

FIG. 6. US nuclear unplanned capability loss factor.

Top quartile capacity factor performance, on a three-year average basis, improved by almost 3 percent from 1997 to 2001 (Figure 7). In contrast, the fourth quartile performance went from 56.4 percent to 82.1 percent, an improvement of more than 45 percent for the same period. The difference among top quartile performances, in the 1997 to 1999 timeframe, was almost 37 percent. In the period 1999 to 2001, the difference between the first and fourth quartiles had decreased to 13.7 percent. This represents a 164 percent improvement in the lowest quartile performance.

During the 1980s, production costs at U.S. nuclear power plants continued to rise (Figure 8). These increases were primarily due to staffing and upward pressures on labor costs. Beginning in 1987, production costs, on cents per kilowatt-hour basis, began to decline. This decline was due to a combination of increased generation, as well as a decrease in production costs. Over the last 15 years, great strides have been made in reducing and controlling plant operating costs, especially labor costs. Numerous U.S. nuclear plants have gone through downsizing efforts and hiring freezes. As a result, the U.S. has seen a steady decline in nuclear production costs on a cents per kilowatt-hour basis.

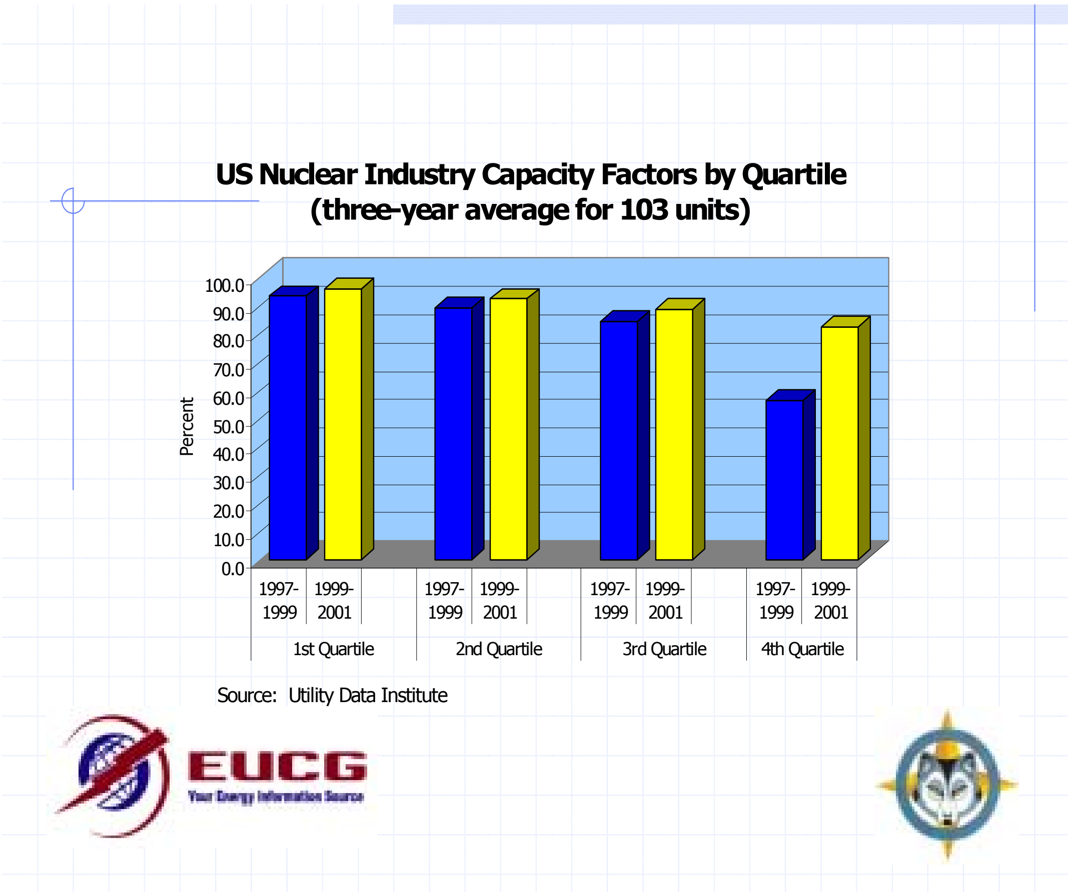

FIG. 7. US nuclear industry capacity facotrs by quartile.

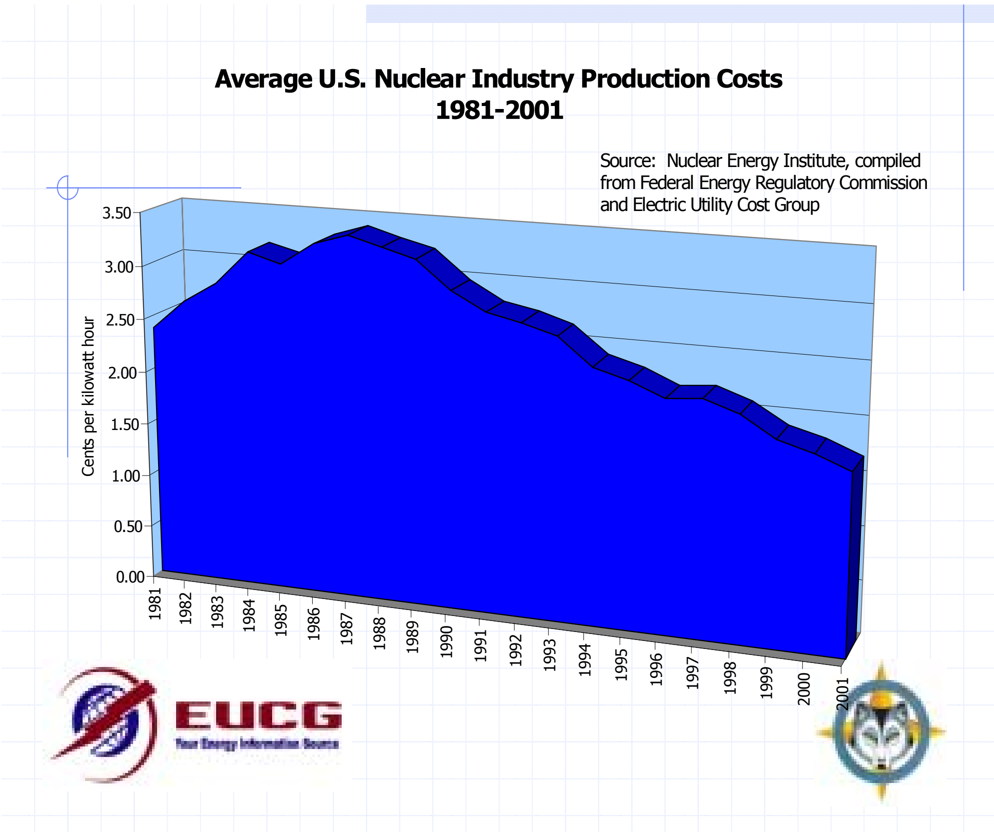

FIG. 8. Average US nuclear industru production costs.

Top quartile production cost performance has improved in much the same way as capacity factor performance (Figure 9). In fact, the improvement in capacity factors has led the way for reducing production cost on a cents per kilowatt-hour basis. The difference between first and fourth quartile performance for the three-year average from 1997 to 1999 was 2.42 cents per kilowatt-hour. The same comparison from 1999 to 2001 shows a difference of only 0.68 cents, an improvement of more than 250 percent. While top quartile three-year average performance has improved by less than 1 percent, fourth quartile performance has improved by 90 percent for the years from 1997 to 2001.

While we have continued to see steady improvement in the economics of U.S. nuclear power over the past decade, our ability to sustain that level of improvement will be challenged in the future. Several factors have and will continue to place upward pressure on U.S. nuclear production costs (Figure 10). These factors include increases in security, materials integrity issues, nuclear property insurance and employee benefit costs.

Our first challenge stems from the increased emphasis on nuclear power plant security in the U.S. Since 2001, more than 2,000 additional security positions have been added to U.S. nuclear plant security forces bringing the total to 7,000 at 67 plant sites. This represents a 35 percent increase in security staffing over 2001 levels. Additional U.S. security-related spending, a 44 percent increase over 2001, is estimated at $370 million for manpower and security-related capital improvements.

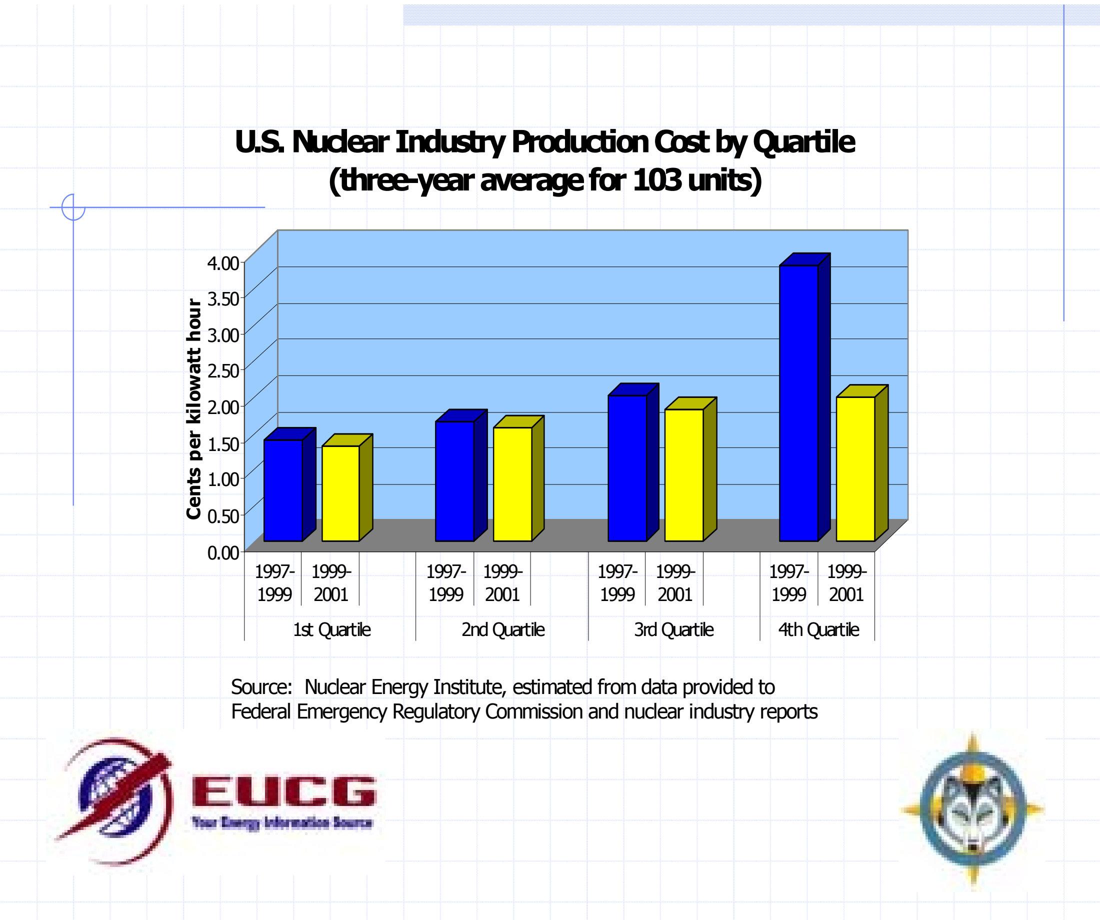

FIG. 9. US nuclear indusstry production cost by quartile.

Materials integrity issues also have become a top priority for the U.S. nuclear industry. Nozzle cracking, boric acid leakage and corrosion have led to increased plant inspections and regulatory oversight. This also has resulted in an increased emphasis on plant safety cultures.

The industry has developed an integrated approach to deal with these issues with the primary emphasis on early detection of material degradation. Other solutions will require costly equipment removal and replacement.An increase in the risk associated with U.S. nuclear power plants also has led to a substantial increase in nuclear property insurance premiums. In the recent past, most nuclear power plants received substantial nuclear insurance refunds and reduced premiums. These refunds were based on sustained accident-free nuclear operations and a low level of perceived risk. After the events in September 2001, the perceived risk at U.S. nuclear power plants escalated. The result—increased property insurance premiums and substantially reduced refunds.

Labor costs at U.S. nuclear power plants continue to account for approximately 73 percent of our total operating costs. Employee-related benefits, such as health care and pensions, have increased dramatically over the past several years. Much of our economic success in the future will depend on our ability to manage our labor costs.

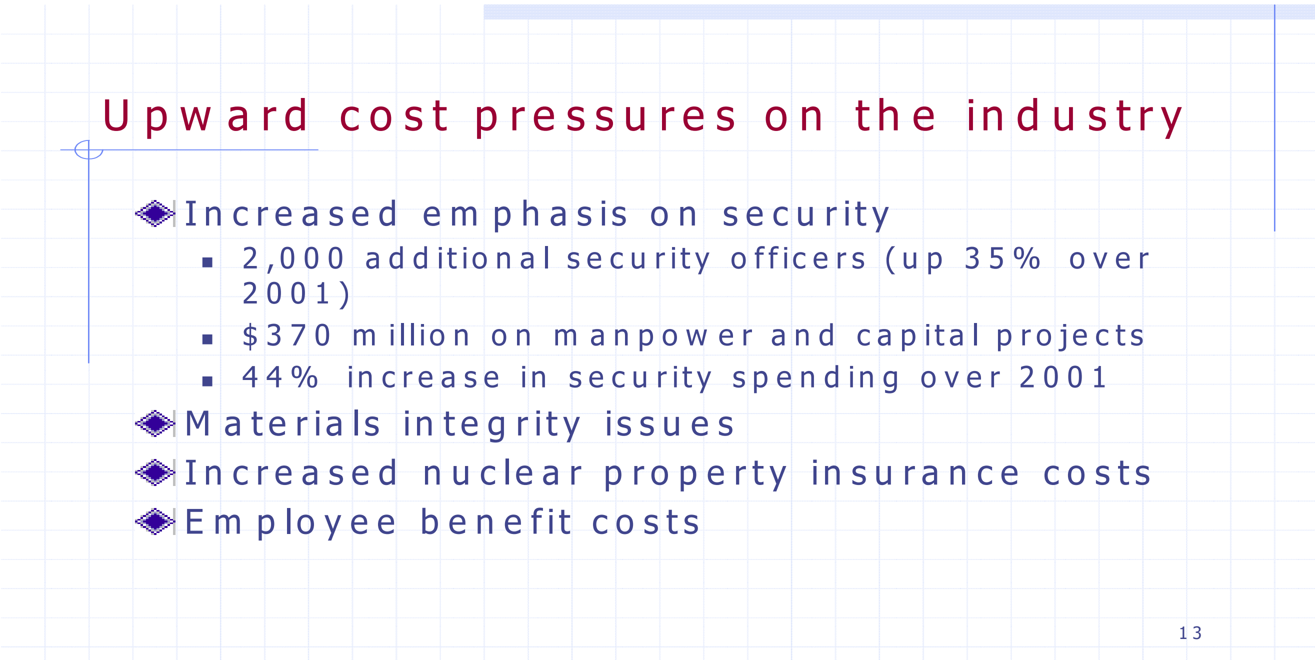

FIG. 10. Upward cost pressure on the industry.

2. CONCLUSION

In the future, the U.S. nuclear power industry will be faced with continued upward cost pressures. Nuclear plant operators will come under increased pressure to manage safety margins as well as managing the bottom-line. As a result, production cost on a cents per kilowatt-hour basis could level off or begin to climb. Future economic survival will depend on the industry's ability to create innovative solutions for controlling costs through process improvements and un-tapping increased generation potential.

INNOVATION IN NUCLEAR TECHNOLOGY AND THE ROLE OF THE IAEA

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Abstract. The picture of nuclear power's role is significantly different within different countries and different world regions. In a few countries large numbers of nuclear power plants are playing a key role in supplying their countries' electricity. The Agency's mandate contains three fundamental objectives: to enhance the contribution of nuclear technologies towards meeting the needs of Member States, to help prevent nuclear weapons proliferation; and to help ensure nuclear safety worldwide. This paper discusses the need for innovative approaches and the role of the IAEA.

1. A NEED FOR INNOVATIVE APPROACHES

Nuclear energy has many attractive features that make it an important part of the world's electricity supply.

Moreover its feasibility as a major source of electricity has been well demonstrated at the national level. Three countries in particular have national nuclear capacities greater than 10GW(e) (Table I), and generate today more than one third of their electricity from nuclear power.

In addition to nuclear power's direct contribution to electricity supplies in these countries, they also enjoy the indirect benefits of nuclear power's contributions to the security of their energy supplies, to their clean air due to very low air pollution from nuclear power plants, and to their efforts to reduce greenhouse gas emissions, due to nuclear power plants' very low emissions of these gases.

Table I. Nuclear energy in three countries

Country	Region	Nuclear Capacity, GW(e)	Nuclear share of electricity generation in 2002, %
France	Europe	63	78.1
Japan	Far East	44	34.5
Korea, Rep of	Far East	13	38.6

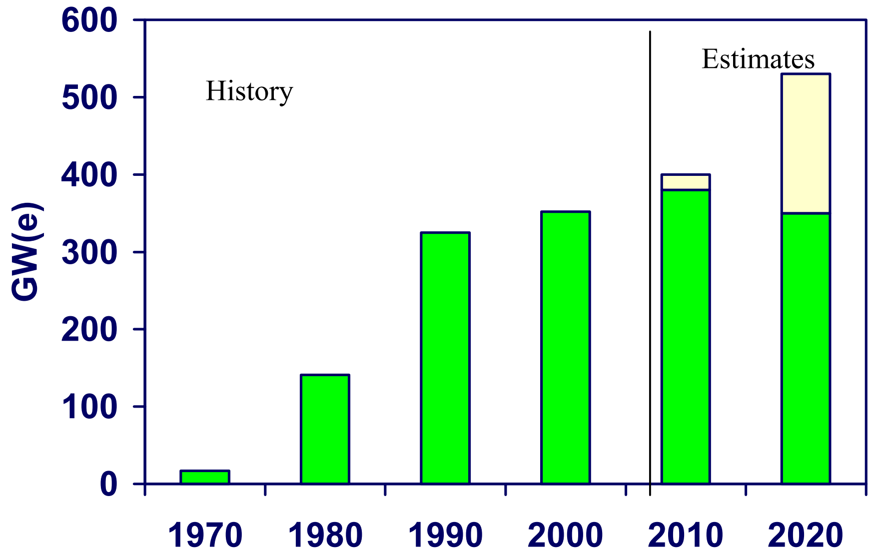

FIG. 1. Global nuclear power capacity.

These national success stories are very important, but for a discussion of nuclear power's wider global potential we need to look more broadly at how nuclear power is used around the world.

Figure 1 shows the historical growth of global nuclear power capacity, in GWe, from 1970 to 2000, and near term projections out to 2020. Figure 2 shows the same sort of historical data and projections for total global nuclear electricity generation in TW·h.

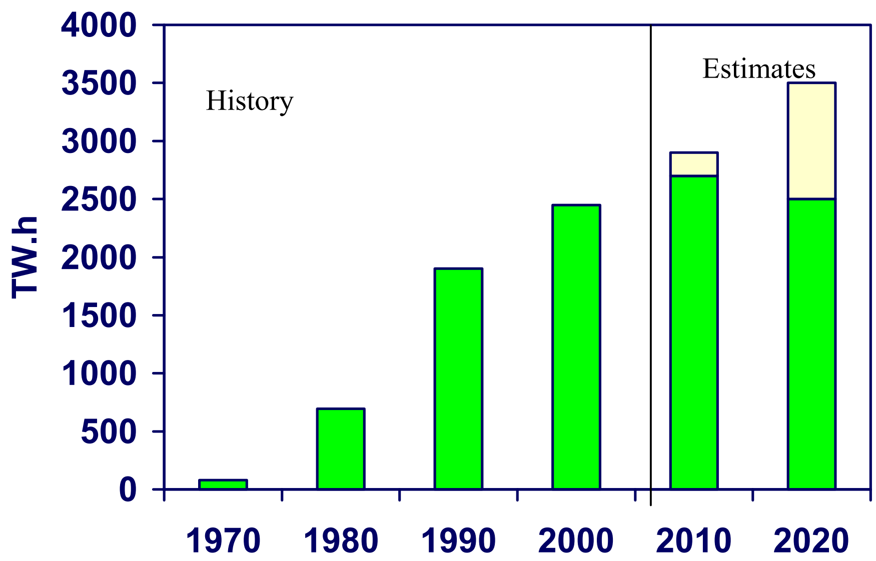

FIG. 2. Global nuclear electricity generation.

As of January 1 of this year, 2003, the total global capacity was 359 GW(e) and the total global nuclear electricity production for the most recent year with complete data (2001) was 2543 TW·h.

In looking ahead, the Agency each year assembles a group of experts to estimate a range of nuclear power expansion possibilities for the next two decades. The future is always uncertain, so we develop two scenarios – a low case and a high case. The most recent low case projects that in 2020 the global nuclear power capacity will be essentially the same as it is today. The most recent high case projects an expansion to 510 GWe, a 42% increase. The total amount of electricity produced by nuclear power is projected to be 2590 TW·h in 2020 in the low case, again about the same as today. In the high case it is projected to be 3710 TW·h, a 46% increase.

In the last two years, a number of observers have raised the possibility of a "new nuclear renaissance" with rapid nuclear capacity expansion in the future that is much more similar to the rapid expansion shown in Figure 1 between 1970 and 1990, than it is to the slow expansion from 1990 to 2000. The Agency's high projection to 2020 corresponds essentially to such a nuclear renaissance.

But even in that high case, how important a contribution would nuclear power make to the world's electricity needs?

Figure 3 shows the growth of nuclear and non-nuclear electricity generating capacity for the world as a whole.

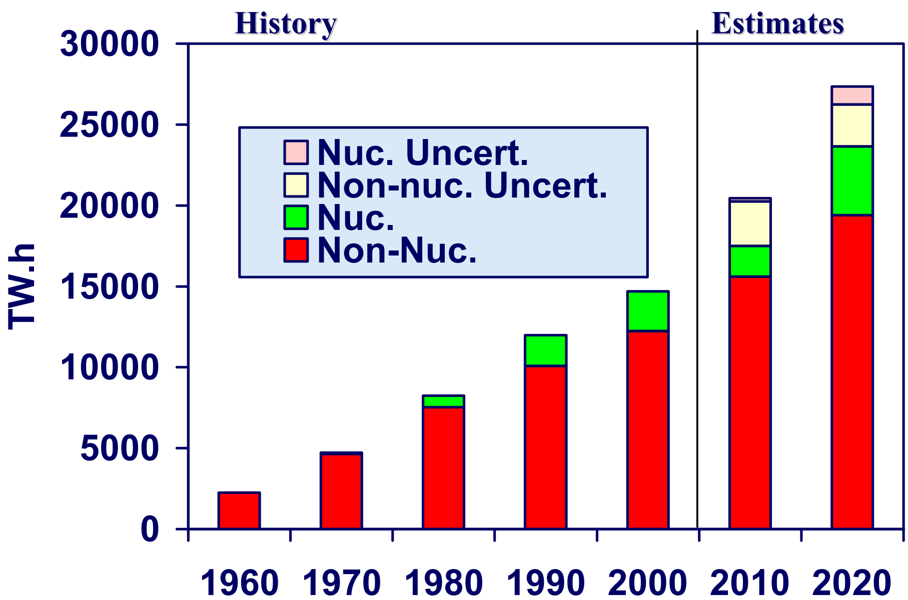

FIG. 3. Global non-nuclear and nuclear electricity generation.

It shows the changing relative global role of nuclear generating capacity (in green) and non-nuclear electricity generating capacity (in red) since 1960 and projected out to 2020 using the Agency's high and lows cases. In the projections in the last two bars, the striped red segments show the range from low case to high case for non-nuclear electricity generating and the striped green segments show the range for nuclear capacity.

Looking at the just the green segments, we see the trend seen earlier in Figure 1 of nuclear capacity emerging in the 1970s and 1980, but then essentially stabilizing during the 1990s with, as indicated in Figure 3, no real expectation of a significantly changed role in the next two decades, even in the high projection case.

What is also interesting in Figure 3 is that the uncertainties in projected non-nuclear electricity capacity, particularly in 2020, are just by themselves *greater* than the total global nuclear capacity, even in the high case.

Figure 4 shows growth of nuclear and non-nuclear electricity generation at the global level. Note that in contrast to the growth of nuclear *capacity* shown in the last figure, nuclear *generation* as shown in this graph increased significantly during the 1990s despite almost no growth in nuclear capacity. The reason is the steady growth in the effective world average capacity factor during this period. But the main conclusion from this graph is the same as from the previous one. Even in the high case, total nuclear electricity generation in 2020 is less than just the uncertainty in the projected non-nuclear electricity generation.

Figure 5 shows the changes of nuclear power's relative role in global nuclear electricity generation. The maximum share occurred in 1995 when nuclear power provided 18% of the total global electricity supply. That share has since declined to 16% today. And our projections are that it will continue to drop. Even in the Agency's high projection, nuclear power's share of global electricity generation decreases to 14% by 2020, and in the low case, it drops even lower to 12%.

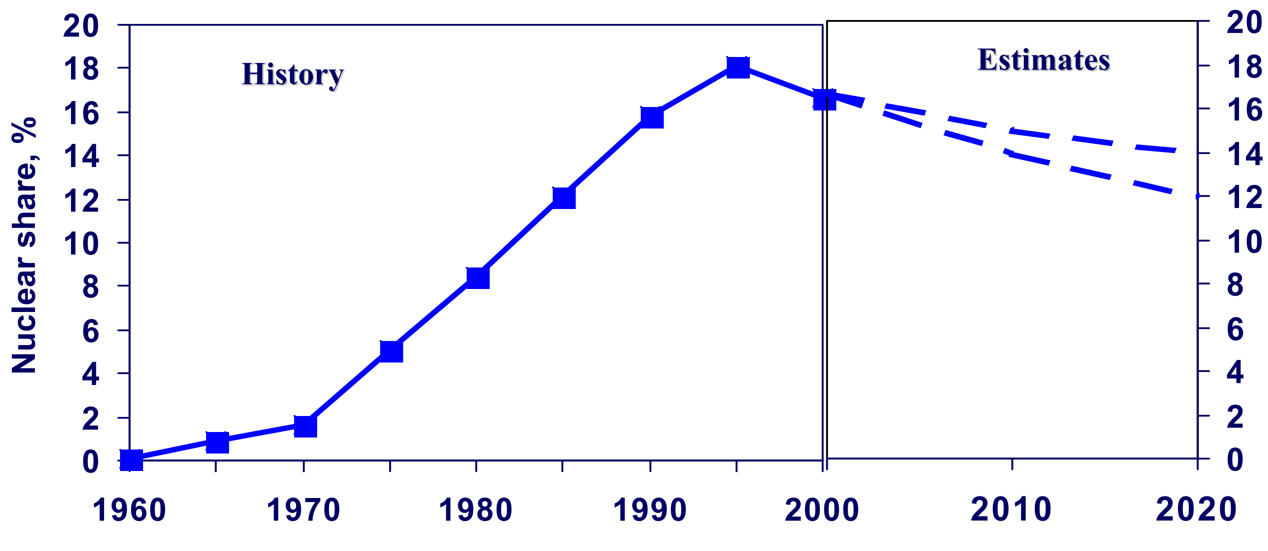

FIG. 4. Nuclear share in global elec. generation.

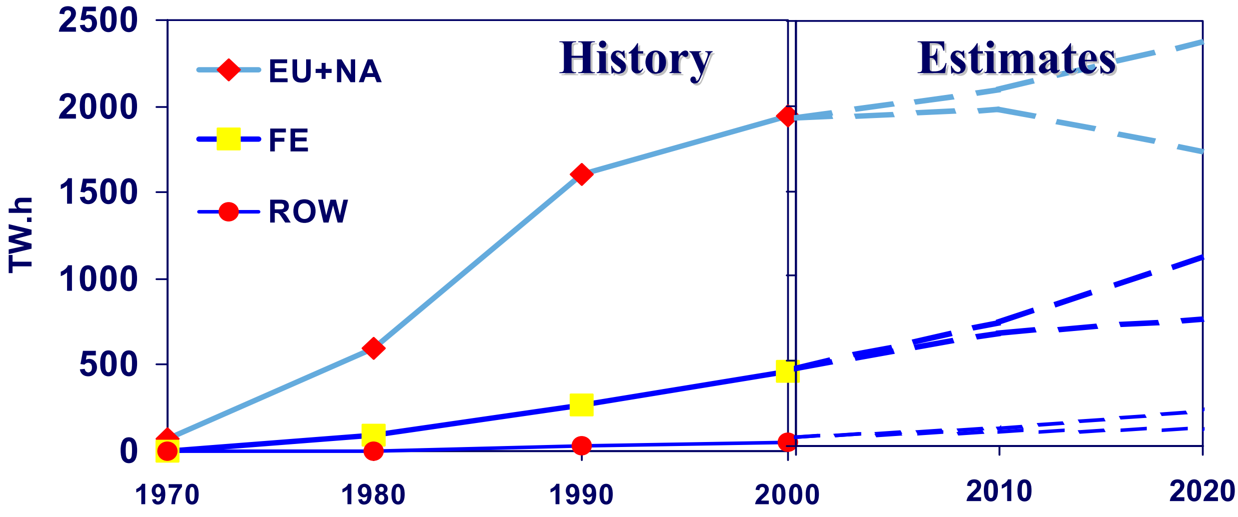

FIG. 5. Nuclear electricity generation by regions.

So far the status and trends of nuclear power entirely was presented at the global level. However, developments are not uniform around the globe, and it is important to understand the different regional trends that go into making up the overall global picture. The Agency collects and presents nuclear and non-nuclear related data for eight different regions of the world. Here, for simplicity, we combine regions with similar trends into three aggregated regions, which then display distinctively different nuclear energy development dynamics.

— The first combined region covers Europe and North America. (EU+NA). Its population is about 1.2 billion, or some 20% of the world population. The nuclear power capacity installed in this region today generates most of the world nuclear generated electricity (78%) at present.
— The second region is the Far East (FE), including Japan; the Republic of Korea; China; and Taiwan, China. Its population of 1.7 billion is somewhat greater that 25% of the global population. At present this region accounts for some 19% of the global total for nuclear generated electricity.
— The third combined region includes everything else – Latin America, Africa and Asia. Its population is 3.2 billion, or more than 50% of the world's population, but it share of the world's total nuclear power generation is less than just 3%.

Figure 6 shows the trends of nuclear electricity generation within these three combined regions. It shows increases in nuclear electricity generation in all three regions, but with some important differences.

In the first region, Europe plus North America, we see substantial increases in nuclear power in the 1970s and particularly the 1980s. In the 1990s growth continued, although not as fast as previously. I should note that in the 1990s nuclear capacity additions were negligible in this region, but there is still a significant increase in nuclear electricity generation. Again this is due to important increases in the average capacity factor for existing reactors in the region. Figure 6 shows that in both the high and low projections, nuclear electricity generation is expected to grow through the current decade, to 2010. Subsequently, the high case projects further increases while the low case projects a decrease to, by 2020, below even today's level.

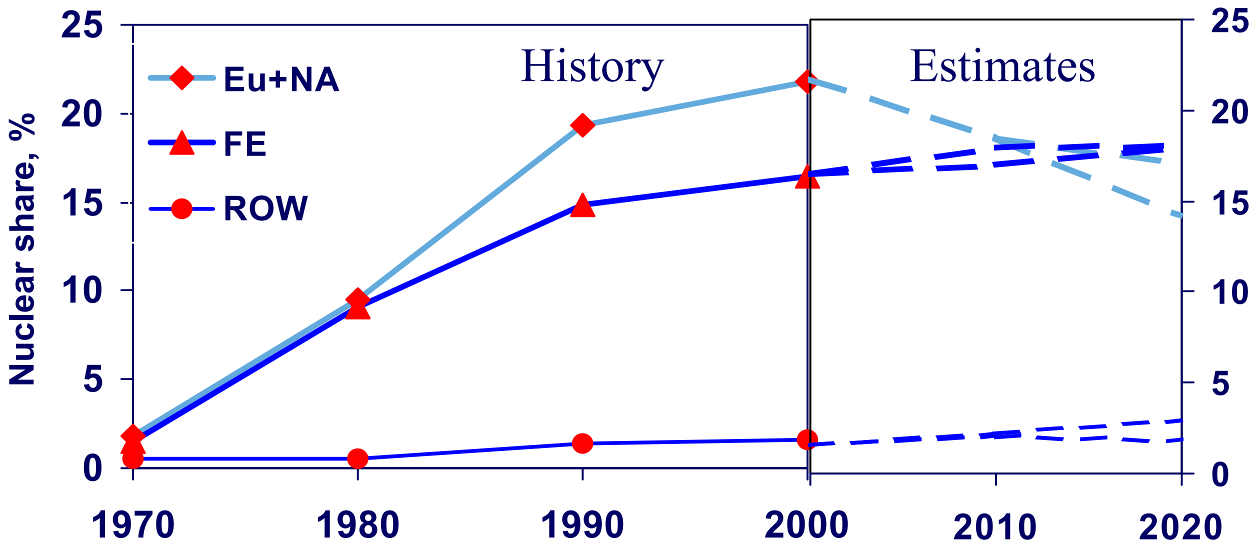

FIG. 6. Nuclear share in electricity generation by regions.

In the Far East region growth of nuclear generation was less intensive in the 1970s and 1980s than in Europe and North America. But it was steady and is expected to continue fairly steadily for the next two decades. The projected range is from slightly faster growth than in the past in the high case to slightly slower, but still positive, growth in the low case.

As for the third combined region, the Rest of the World, the picture with nuclear use is a sobering one. Nuclear has not played any notable role up to the present and there is no expectation of significant changes in coming decades.

Figure 7 shows trends in nuclear power's share of the electricity market in each of the three aggregated regions. Up until today nuclear's share has been increasing in all three regions. The most significant growth has been in Europe and North America, where nuclear generation has grown continuously and now provides 22% or the region's electricity.

In the 1970s, for the Far East, the growth in nuclear's share of the electricity market followed practically the same trend as in Europe and North America but then slowed, mainly because of China, which became much more important in the region's overall electricity generation in the 1980s and 1990s but relied mostly on non-nuclear electricity. The present share of nuclear electricity in the region today is some 16%.

In the Rest of the World, nuclear's share of total electricity generation has grown but is still below 3%.

For the next two decades there are important differences in the projections for the three regions. In the Far East the slow steady growth in nuclear's share in the 1990s is expected to continue and bring it to 17-18% in 2020. In the Rest of the World nuclear's share is projected to be essentially flat. The most dramatic departure from the past is projected for Europe and North America, where nuclear's share of electricity generation is projected to reverse over the next two decades and drop to 17% in the high case and 14% in the low case.

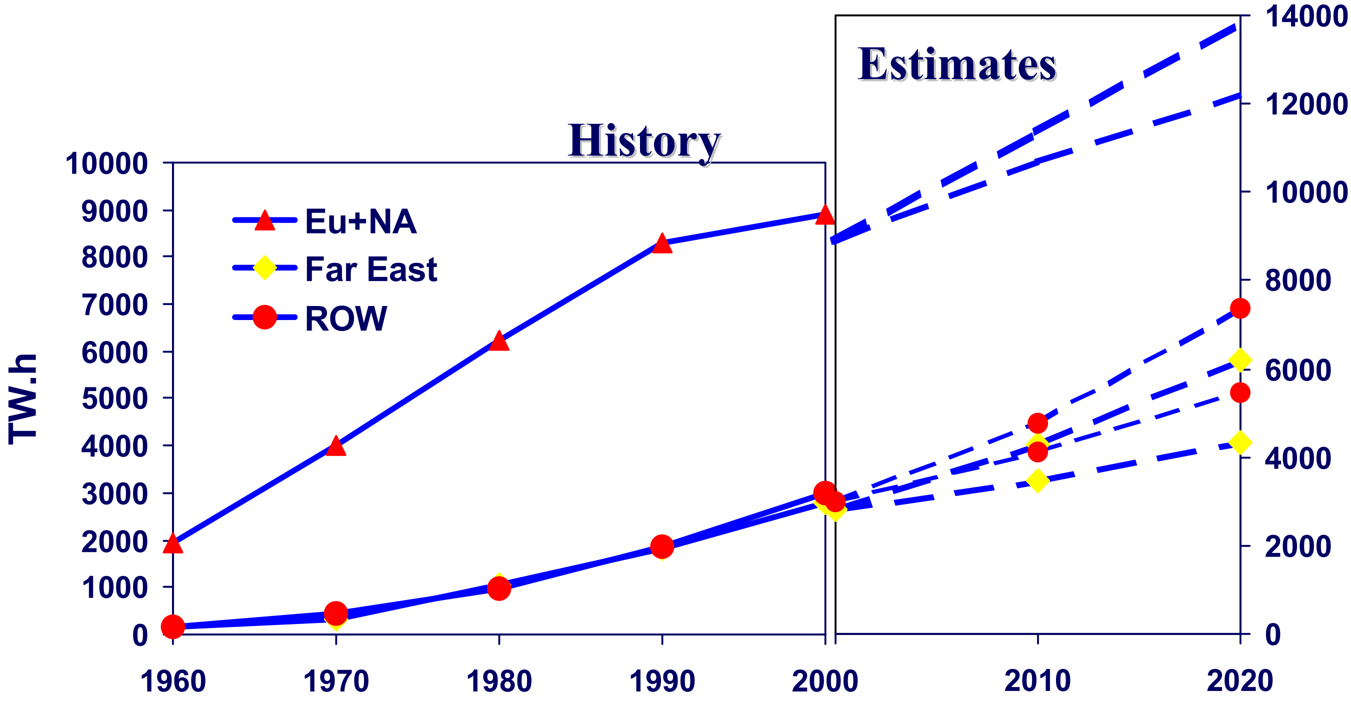

FIG. 7. Total electricity generation in regions.

These are trends for the market share of nuclear power in overall electricity generation. To get the full picture, we should also look at how the overall electricity market is changing in each of these regions.

Figure 8 shows the historical and projected growth in overall electricity generation in each of the three regions. Europe and North America has been the biggest electricity consumer, and will continue to be for the next two decades. Electricity generation is expected to continue to increase in this region at roughly its pre-1990 rate, but with significantly less reliance on nuclear power as we saw in Figure 7. The expected preference for new generating capacity in the next two is for natural gas. Because this region accounts for the bulk of the world's nuclear generation today, when viewed from the global perspective shown in Figure 5 its projected decline in nuclear power's share of the electricity market outweighs the projected modest increases in the Far East and the Rest of the World.

The second cause of the projected global drop in nuclear's share of electricity generation is the growing contribution of the Rest of the World to global electricity consumption, while still having a very low share of nuclear electricity, much lower than the other two regions.

Summing up:

— The picture of nuclear power's role is significantly different within different countries and different world regions. In a few countries large numbers of nuclear power plants are playing a key role in supplying their countries' electricity

— The global picture, however, it is not very encouraging. Nuclear power's share of global electricity generation reached its maximum of some 18% in 1995 and then started to shrink. It is 16% today and is expected to continue falling down to between 12% and 14% by 2020. This would put its total contribution within just the uncertainty band of the level of non-nuclear electricity generation projected for the same period.

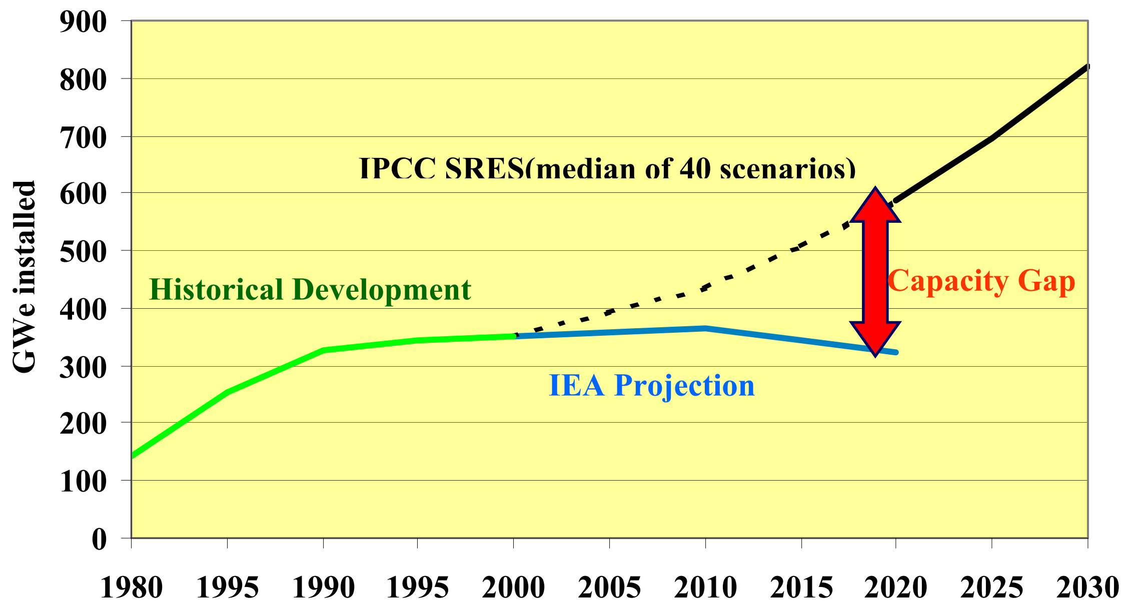

FIG. 8. Electricity generation in regions.

What are the main causes behind this picture? They are different in the different regions.

In Europe and North America, the growth of nuclear capacity faces mainly economic challenges arising from electricity market deregulation, plus improvements in particularly gas-fired power generation. In some countries political and public concerns over nuclear waste disposal or safety issues are also hindering the use of nuclear power.

In developing countries, the main issues limiting nuclear expansion are a lack of expertise in nuclear technology and the necessary safety culture, a lack of adequate infrastructure, and of course economic and financing issues. Waste management and non-proliferation issues are additional obstacles in these countries.

To address all these challenges, the nuclear community continues to work on a number of fronts.

— Recent decades have seen record improvements in the aggregate availability of nuclear power plants. The trend is significant and substantial. However, this route to success has its limits as will necessarily level out somewhere around 85% or 90% over the coming decades.
— Extending licensed lifetimes and uprating existing nuclear power plants are other very important steps that are occurring more frequently and will continue in the coming decades. But again these routes to expansion have their limits.
— The impressive health and environmental advantages of nuclear power, relative to alternative electricity generation options, have been analyzed and advertised. However, to the extent that the corresponding disadvantages of fossil-fuel alternatives are not internalized in capital and operating costs, nuclear's health and environmental benefits have little impact on investors in new generating capacity. They prefer the cheapest option, and most often it is not nuclear.
— The industry continues work on evolutionary new designs to improve NPP performance even further. New designs for advanced LWR reactors may well have a market in a

limited number of countries or regions that do not have access to cheap gas from pipelines or coal.

In short, the nuclear community has accomplished much, but evidently not enough yet to turn around the trends shown in Figure 5.

If you look at longer term scenarios, such as those published in 2000 in the Special Report on Emissions Scenarios (SRES) by the Intergovernmental Panel on Climate Change (IPCC), most of them project nuclear capacity growth above that in our high estimate by 2020, with continuing growth beyond 2020 all the way to 2100. There is thus something of a "projection gap" between the modest near term projections and the more optimistic long-term scenarios.

Closing the projection gap requires success on two fronts. First, progress must be made on political and public acceptance issues. This will require continuing public and political discussion of the pros and cons of all energy options.

Second, through evolutionary and innovative improvements, the nuclear industry must continue to reduce costs and improve safety, waste and non-proliferation features of nuclear energy systems. This will require continuing innovative R&D on the part of industry and governments. If the nuclear power sector is to increase its role significantly – to 30-50% of the global electricity market for example – it cannot simply continue to do what it has been doing and expect that factors outside its control, such as fossil fuel prices or environmental taxes, will change to make nuclear power's prospects more favorable. To reach a different outcome than that indicated by current trends, something must be done within the nuclear community to generate new technological solutions. The challenge is to look to the future, to identify what innovations and new directions – that build upon and make good use of existing expertise and accomplishments – are most promising for helping nuclear power capture a *growing* share of a growing market. More of the same won't do. The industry must look to the future and must be innovative.

2. THE ROLE OF THE IAEA

The Agency's mandate contains three fundamental objectives: to enhance the contribution of nuclear technologies towards meeting the needs of Member States, to help prevent nuclear weapons proliferation; and to help ensure nuclear safety worldwide

Many of our Member States are carrying out R&D related to nuclear innovation. What then is the Agency's role?

The main challenge facing us today is not in itself the R&D carried out in many countries over a great many years, but rather in trying to ask ourselves what <u>kind</u> of R&D would be most effective. This challenge requires an international response

Many nuclear challenges have international dimensions that can be understood and addressed properly only at the international level, e.g. international safety requirements, international spent fuel movement and waste management, and international safeguards and security measures.

We have to ask ourselves what role nuclear energy might have in the future. And in what regions of the world and to what extent nuclear energy could be needed in the future. In the context of these specific questions, we are thinking in terms of at least up to 2050, which is the period of time when innovative technologies be expected to dominate the world's nuclear energy system. It is essential, however, that we also understand the regional differences, as was shown in first part of the *paper.*

We came to realize that the Agency was the only international organization where these specific issues could be discussed and addressed.

To this end the IAEA launched a new International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) — a major extra-budgetary initiative coordinated by the Agency — in September 2000. INPRO invited all interested Member States, both technology suppliers and users, to consider jointly international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles.

INPRO is looking ahead to what the market might be for nuclear technology around mid-century. Are people more likely to want big plants or small plants? How big will differences be among regions, as well as among industrialised and developing countries? What kind of infrastructures of countries and regions, and even international infrastructures would be necessary for deploying innovative nuclear concepts and fuel cycles? What are the legal and institutional requirements; education, training and R&D implications and socio-political implications in the case of innovative reactor development? Will nuclear energy still be used almost entirely to generate electricity, or will it be used significantly for industrial heat, to make chemical fuels that can be used in the transportation sector, or to desalinate seawater to produce fresh clean drinking water?

A range of answers has to be examined and made available to policy makers designing national R&D strategies. Based on the same information, different countries may then choose different strategies.

Our job is to bring experts together from the full range of participating countries to learn how others see the future, and to create an international global perspective that is greater than the sum of the contributing national perspectives.

The preliminary results of the INPRO project have been presented at various international conferences to encourage feedback. The final draft of the INPRO Phase 1A Report was approved by Steering Committee in May 2003. Activities for Phase IB will start with case studies, to provide feedback on Phase 1A's user requirements and methodology.

Upon successful completion of Phase I, taking into account advice from the Steering Committee, and with the approval of participating Member States, Phase II of INPRO may be initiated. Drawing on the results from Phase I, it will be directed to:

— examining in the context of available technologies the feasibility of commencing an international project;
— identifying technologies which might be appropriate for implementation by Member States participating in such an international project.

There is the possibility of organizing crosscutting research in technical areas related to both the generic system concepts selected by INPRO Member States.

Also there may be an interest in organizing CRPs in the areas of enabling technologies that may be of interest INPRO Member States, including many developing countries.

It is also important to mention here that, in the potential areas of R&D activities which I have just mentioned, the Agency would not be starting from scratch. The Agency has a solid background here.

To catalyse innovation my own department, Nuclear Energy, supports the development of new reactor designs and fuel cycle technologies in Member States through coordinating analyses and fostering information exchange in the framework of several Technical Working Groups on advanced nuclear technologies.

The Agency similarly supports improvements, applications and dissemination of global knowledge on all aspects of the fuel cycle through the Agency's databases, technical documents, guidelines, research projects, peer reviews, and training and informational meetings.

The principle objectives of many innovative designs are achieving significantly better economic, safety, waste, resource and non-proliferation performance in comparison with existing reactors and fuel cycles. There are also under development new designs for non-electric applications such as desalination, district heating and hydrogen production.

The Agency is promoting information exchange activities in the area of fast reactors. Development objectives for fast reactors now include not only increasing the efficiency of uranium resource utilization but also incinerating plutonium stocks and reducing long lived radioactive waste for disposal. Fast reactors and accelerator driven systems are being developed in some Member States as a possible response to the challenges of long term waste storage and potential proliferation risks. The Agency provides and facilitates information exchange and collaborative research and development, fostering the pooling of resources and expertise.

To address issues associated with the development of advanced or innovative nuclear fuel cycle technologies the Agency organized a Technical Working Group on Nuclear Fuel Cycle Options. This group addresses, for instance, the management of accumulated fissile materials, including separated plutonium, the development of advanced fuel cycle concepts with enhanced proliferation resistance features, the effective management of spent fuel, etc.

Modular high temperature gas cooled reactors with coated particle fuel, inherent safety features and passive systems, coupled with state of the art power conversion technology have sparked renewed interest. They can be used for more efficient electricity generation, co-generation of electricity, and heat and high temperature applications, particular for hydrogen generation. The Agency's 'Knowledge Base' web site on HTGR technology continues to draw global attention as a source of information and publications. In 2001 the Agency issued a status report on HTGR activities.

The International Nuclear Desalination Advisory Group (INDAG) has recently reviewed nuclear desalination activities, evaluated the Agency's programme in this area and proposed potential new activities for our 2004–2005 programme to accelerate the deployment of nuclear desalination projects. Progress on the Agency activities in nuclear desalination was reported to the General Conference in September 2002, which approved a resolution requesting a continuing and strengthened programme on nuclear desalination using small and medium sized reactors.

3. CONCLUSION

Today, as in the time of the nuclear pioneers, civil nuclear energy faces a paradox. Its advantages and its potential to play a key and growing role in the global energy supply are clear and obvious. But it faces near-term important obstacles that could cause much of its potential to go unrealized.

Today – just as at the beginning of the nuclear era – we must focus on these obstacles and address them efficiently. Development of a global vision on nuclear energy potentials and challenges in a new changed political and economic environment becomes a crucial task for the nuclear community. Activities in these directions have started within the INPRO project, as its first phase aims at the definition of nuclear scenarios and possible user requirements for a new generation of reactors and fuel cycles. The work we have started in this area has shown these to be very challenging tasks that require international participation from both the major producers of nuclear technologies as well as the potential users of these technologies. It is clear that innovation will be essential, and our objective is that the INPRO project should make an important contribution to this process. We hope that its global character, encompassing both designers and end users and their user requirements, its long time horizon, its consideration of the changing energy sector and its broad based input through IAEA membership will all make it a valuable forum for resolving the above challenges.

EVOLUTION OF TECHNICAL, SOCIAL, ECONOMIC AND POLITICAL CONDITIONS

(Session 3)

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WHAT DO SOCIETIES EXPECT FROM INNOVATIVE NUCLEAR TECHNOLOGIES ?

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Abstract. There are no universal views on nuclear power, nor universal expectations about its future progresses. Therefore, this paper presents only one view, mostly based on European experience. There are misconceptions and misrepresentations about the present status of nuclear technology. In that context, the perceived risks associated with radioactive wastes management are generally overblown. It would appear that the first expectation from innovative reactors is the guarantee of no significant release of radioactivity in the environment under any circumstances. The second expectation concerns almost unanimously the radioactive wastes : less waste, less long-lived wastes, and no waste at all if possible. This request for "no waste" has been exacerbated by some exaggerated claims by the proponents of such or such reactor design. Expectations about non-proliferation vary vastly from place to place : in some countries it is high in the public agenda, in other countries, it is a non-issue. Making a better use of the mineral resources is a potent motivation for innovation among the nuclear technologists, but it is no longer and not yet a public concern. But there are two last and rather overwhelming expectations which put the other in perspective : reliable power supply and reasonable electricity prices.

1. INTRODUCTION

In a Finnish cartoon that I love, a character declares :"There are two things which bother me about nuclear power : I know absolutely nothing about it and I distrust those who know".

It would be rather presumptuous to pretend answering with any degree of certainty or accuracy to the question : "What do societies expect from innovative nuclear technologies ?", because societies are very diverse, and they more often express frustrations than expectations, when they express anything at all. This paper should therefore be considered only as one view of the question, supported by analyses of polls carried out in several countries – but which seldom asked the question in those terms – and mostly based on a long personal experience of interacting with the public in many different fora, and in several regions of the world.

First of all, expectation depend upon the perceptions of the benefits and drawbacks of nuclear power as it is implemented today, and there are misconceptions and misrepresentations about the present status of nuclear technology. To list a few :

— A majority of people believe that nuclear power contributes significantly to global warming and climate change. This is vividly illustrated by a "Eurobarometer" poll carried out at the end of 2002 in all 15 countries of the European Union (Figure 1). This misconception is very important because public opinions are more and more aware of the threat of climate change, and nuclear energy proponent would put very high on their list the non GHG emitting quality of nuclear power.

— Many are convinced that radioactivity has insidious and mysterious ways to percolate through matter and contaminate the environment. The basic facts that radioactivity can easily be detected and measured, and that it is easy to shield oneself from radiations have been forgotten, if ever learned in the classroom.

— For many people, the longer the "life" of a radioactive element, the more dangerous it must be (without any extrapolation to the eternal life of stable elements). When people are told that Iodine 129 has a "life" of 16 million years, they are scared instead of thinking how little radioactivity this element emits.

— It is very hard to convey the notion that the health effects of radiations depend only from dose and dose rate, irrespective of whether it is "natural" or "artificial" radiation. In that context, the perceived risks associated with radioactive wastes management are generally overblown.

And the list could go on and on. On the other hand, extended recent surveys of French opinions about nuclear power exhibit encouraging features : People believe that nuclear power decreases our dependency from oil and, therefore, the vulnerability of our energy supply. The French know that operating nuclear reactors generate competitive electricity (but they are less affirmative about future reactors). They love Wind and Sun (though the NIMBY syndrome affects significantly the deployment of wind turbines in some regions), but they are aware of their intermittency and do not trust them to really replace nuclear power. And finally, appoint which is crucial for today's topics, they still believe in technical progress and more or les trust us, nuclear scientists and engineers, to devise better mousetraps for to-morrow.

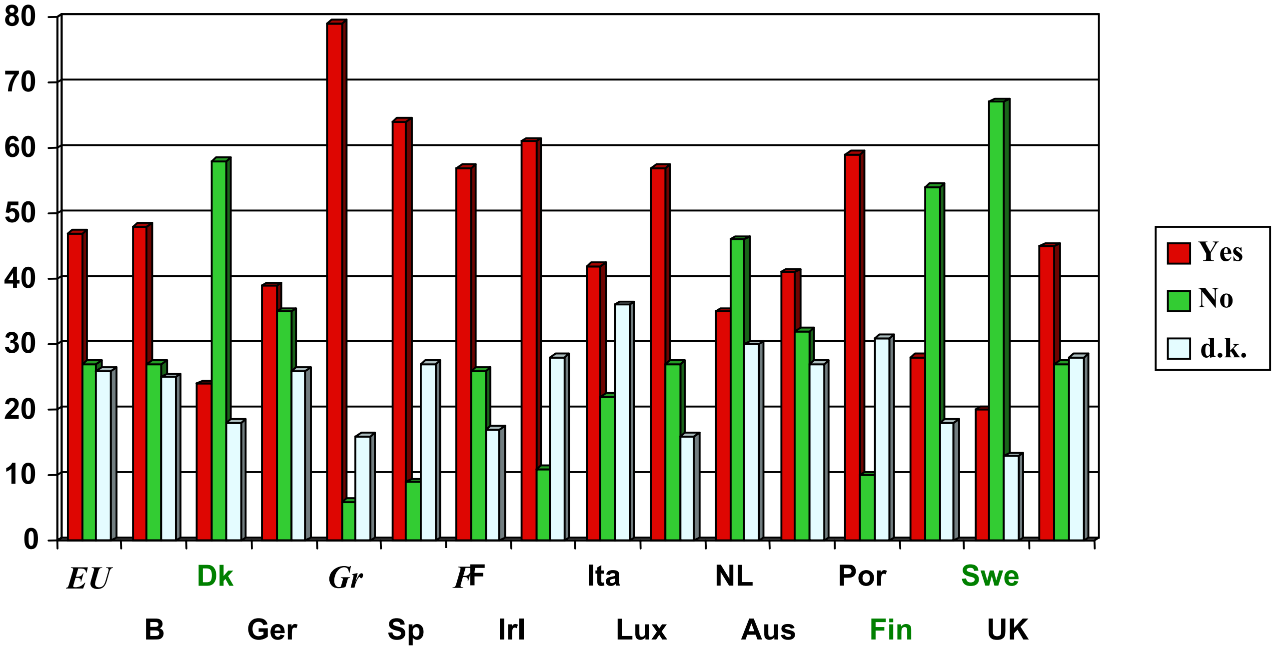

FIG. 1. Nuclear power contributes significantly to global warming and climate change, EUROBAROMETER Dec.02

2. SAFETY

There are at least two universal expectations. It appears that the first demand from innovative reactors is the guarantee of no significant release of radioactivity in the environment *under any circumstances*. "No severe accident", "no meltdown" are often the formulations used, but the actual demand is : "*no radioactivity*".

The debates about passivity, "intrinsic" safety, "walk-away" or other "idiot-proof" reactor designs are mostly restricted to our specialised circles : what the man-in-the-street demands is to be told :"a new Chernobyl is impossible !". And there, we have a problem, because impossible is not part of our usual vocabulary, but the public will not settle for less.

3. WASTE MANAGEMENT AND DISPOSAL

The second expectation concerns almost unanimously the radioactive wastes : less waste, less long-lived wastes, and no waste at all if possible. There again, let me refer to a "Eurobarometer" poll, this one dated November 2001. Citizens of 14 out of the 15 EU countries would agree, by a large majority, with the statement : "*If all waste could be safely managed, nuclear power should remain an option*". I sorry to say it here in Vienna, but the odd man out is Austrian (Figure 2).

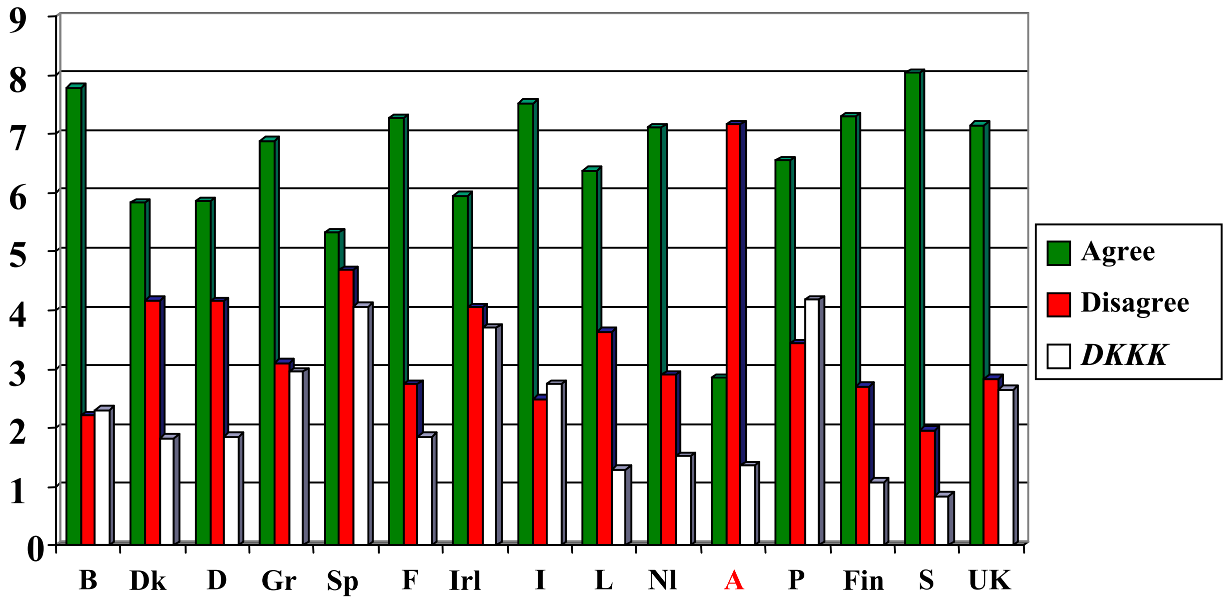

FIG. 2. "If all waste can be safely managed, nuclear power should remain an option", EUROBAROMETER Nov. 2001

The Frenchman in the street does not know what is a radioactive waste, has no idea that there are different categories of wastes with different management practices, but he is nevertheless convinced that radwaste is very dangerous, and that its disposal constitutes an important problem without any solution (it certainly appears that the Finn and the Swede have a better knowledge). This danger, according to some philosophers, is "consciously imaginary".

It is therefore important to convey the message that, **as a matter of fact**, radioactive wastes *are* managed today. They are not orphaned, and the nuclear industry is taking good care of them, under the surveillance and control of the Authorities in charge of Safety and Radioprotection. HLW are safely stored, and cause no immediate danger to anybody. Geologic disposal is technically a suitable solution, and some countries have made significant progress in the last decade towards making it an industrial reality.

But next-generation reactors can do much better to optimise waste production, especially in where long-term radio-toxicity is concerned. They will not, however, make nuclear waste magically disappear, and the public should not be misled on this issue by some exaggerated claims coming from the proponents of such or such reactor design.

Safety and waste management are two universal expectations for innovative nuclear systems, but there are many other expectations with a lesser degree of unanimity.

4. NON-PROLIFERATION

Expectations about non-proliferation vary vastly from place to place : in some countries it is high in the public agenda, in other countries, it is a non-issue. I can offer no explanation for this fact. Non-proliferation is a very important criterion for public acceptation in the United States, but in France, it is mostly a concern, and a serious one, for specialists in nuclear technology and political sciences, as well as a permanent feature of the argumentation of Greenpeace, among others. But it does not rank high among the public worries.

This difference of sensitiveness is not recent, and you could easily track it back to the INFCE times, in the 70s. It might be based on different perceptions of the relative importance of technical and political barriers to proliferation. But I repeat I do not know for sure. As a result, the weight of non-proliferation criteria in the public expectations about new innovative nuclear systems is quite variable.

5. OTHER EXPECTATIONS

In the aftermath of the first oil crisis, many countries were planning aggressive development of nuclear power. Faced with such figures as 2000 GWe in 2000, there was a widespread fear that uranium scarcity would not be far away. We know today that such future was not to be, often forgetting that it might have been and that it will probably happen… later. *Making a better use of the mineral resources* is therefore a potent motivation for innovation among the nuclear technologists : this is no coincidence if 4.5 out of the 6 "targets for R&D" selected by Generation IV International Forum GIF last year were fast neutron reactors. But for the general public it is no longer, and not, yet a nagging concern.

There is a periodic request for *small and medium power reactors*, and the Agency organises regular meetings to update the status of offer & demand for SMPRs. Though it appears to make a lot of sense, we must admit that economics, and the lack of demo-plant in the vendor countries, have discouraged customers until now – at least for reactors fully dedicated to electricity generation.

On the other hand, many societies would favour *dual-purpose* nuclear reactors, for power and heat, or for power and desalination. There again the Agency plays a lead role in keeping the option open. Further along that road, *hydrogen production* is certainly a very popular topic. The geopolitics of oil and the importance of the transportation sector in the increase of GHG emissions combine together to make hydrogen look good. Many people would mention hydrogen among the clean and renewable energy sources, even though it is not an energy source but, rather, an energy vector.

6. SOCIETIES EXPECTATIONS AS REFLECTED IN THE CURRENT PROJECTS

Both INPRO and Generation IV have identified criteria to be met by future innovative systems. All of them reflect some societies expectation, but sometimes with different priorities. As an illustration, I offer my own lecture of the 6 GIF models (Table I).

7. IN LIEU OF CONCLUSION

In all my development, I have carefully avoided two of the major expectations of the public, but I cannot conclude without mentioning them, because they are *sine qua non* conditions for innovative nuclear systems to be part of our sustainable future. These two rather overwhelming expectations, which put the other in perspective, can be thus formulated :

— The first one is : "I expect power when I flip my switch",
— The second one is : "Do not increase my electricity bill "

Table. I. The 6 GIF models

Acronym	Type	1st priority	2nd priority
SFR	Sodium cooled Fast	Fissile economy	Waste minimization
LSF	Lead Alloy cooled Fast	id	id
GFR	Gas cooled Fast	id	Safety
VHTR	High temperature Thermal	Safety	Hydrogen
SCWR	Supercritical water Fast/Thermal	Fissile economy	Thermal efficiency
MSR	Molten salts Thermal	Fissile economy	

TECHNICAL ASPECTS OF INNOVATIVE NUCLEAR SYSTEMS INCLUDING RELIABILITY AND SAFETY

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Abstract. The current stage of innovative technology selection requires extensive interaction between the regulators and engineers, designers, environmentalists, investors and politicians in order to develop mutually acceptable underlying principles, approaches and, if possible, unified standards. New regulatory standards should ensure that safety enhancements precede the increase in a scale of nuclear power development. Preliminary analysis shows that the new technology - the use of solid coolant in nuclear reactors - could meet broad range of User Requirements for innovative technologies. It seems that development of innovative technologies requires a concerted effort of the world community.

1. INTRODUCTION

Accidents at Three-Mile Iceland and Chernobyl have shown the insufficient level of nuclear power safety.

Two important thesis's were put forward:

— nuclear reactors of the future must differ in principle from today's reactors (Alvin Weinberg);
— nuclear technology shall be the "forgiving" one, i.e. a single operator error should make severe accident consequences highly unlikely.

However, designs that were produced or developed in a next 20 years were not characterized by the principle modifications. The safety enhancements were achieved by increasing the number of safety systems that, correspondingly, makes the nuclear facilities more expensive. It has and still is making nuclear power less competitive.

The decrease of nuclear power competitiveness, public concern with regard to nuclear accidents, the lack of acceptable technologies for nuclear spent fuel and waste management resulted in decrease of public confidence followed by decisions taking to reduce the nuclear power use in Sweden and Germany and slowing down in deployment of nuclear power in other countries.

The willing to find a way out of such a situation is clearly defined by Mr. M. El Baradei, the present IAEA Director General: "In my view, a solution to this dilemma may depend heavily on the development of new, innovative reactors and fuel cycle technologies. To be successful, the new technology must be absolutely safe, proliferation resistant and economically competitive."

Recognizing the need to make radical solutions, the IAEA came to a decision to develop the "User Requirements". The final first version of those Requirements became available in May of this year.

"User Requirements" mean that innovative nuclear reactors and their fuel cycles shall meet the advanced standards related to safety, environment protection and radioactive waste management and to be more cost-effective.

The "User Requirements", however, do not set forth any safety conditions that could be interpreted as being more stringent than those contained in the best examples of currently existing reactor and nuclear fuel cycle technologies.

2. REGULATOR'S ROLE IN SEARCH OF INNOVATIONS

The word "User" in the IAEA terminology has a broad meaning. It includes investors, designers\engineers, utilities, Regulatory Bodies, and consumers of nuclear power industry products. Therefore, one can see the great divergence of interests among the members of this community.

Formulation of "User Requirements" will facilitate broad-based international consensus among different User categories regarding acceptance of new technologies, enabling the Users to compare them with each other as well as with the existing technologies.

The Licensing Authorities play specific role as spokesman of combined requirements of community, therefore they might become a key persons in search of optimum correlation for all requirements and criteria.

The representatives of Licensing Authorities together with designers, engineers, fabricators, economists and environmentalists should participate in discussions of innovative technology proposals from the very start. It will allow them to make their recommendations at the benefit of those technologies that should be further developed.

It seems reasonable to discuss two topics, namely:

— International co-operation of Licensing Authorities that might facilitate improvement of "licensing infrastructure" and licensing activity, and
— Development of mutually acceptable underlying principles, approaches and, if possible, unified standards to be applied to innovative technologies at the international level.

The developing countries where nuclear power will be developing at a faster than average pace could become directly involved in this collaboration.

Regulatory Authorities are responsible for formulation, approval and entering into force of Regulatory Requirements.

3. THE INCREASING SCALE OF NUCLEAR POWER INDUSTRY CALLS FOR CORRECTION OF REGULATIONS

Regulatory Requirements change with the development of society and technology. Predicted deployment of nuclear energy application shall not be accompanied by increase of probability of radiological impact. Requirements related to significant radioactive release probability should be tightened at a faster rate than the rate of growth in the nuclear industry.

Since in reality the nuclear installations of the old and innovative design will have to coexist for a rather long time, this increased stringency will naturally have a greater impact on innovative projects.

Such logic leads to the necessity to assume as the innovative design only those reactors in which the considerable radioactive releases are intrinsically impossible and brought to negligibly small amount while considering all force-majeure external circumstances.

The same requirements relate to the external fuel cycle as well, assuming also that in this case one should consider not only the emergency, but also the routine releases\discharges, mainly to the air and water. The release of long-lived Alpha-sources such as plutonium shall be considerably reduced taking into account the development of radiochemistry and cumulative effect of discharges..

Review of the recently received innovative technology proposals demonstrates once again the complexity of generating truly innovative ideas. A lot of interesting and useful proposals available represent the logical development of existing technologies that satisfactory proved from the today's position. The more practical experience there is in the use of economic and safety indicators, the more confidence inspire.

This logic points us toward the evolutionary path of nuclear power development. The "Guidelines of INPRO" provides the reactor types that characterized by the wide scope of experience of use (such as reactors with light water) or, at least, by the long development history (as the high-temperature reactors and breeders with liquid-metal coolant).

These Projects have the advantage of better understanding of processes and the higher degree of assessment reliability, however, unfortunately, they have also the predetermined limits of economic and safety indicator improvement.

We appreciate very much the "principles" and "requirements" described in the "INPRO" but assume that two matters should be emphasized. For the broad development of nuclear power industry, besides of reproduction of nuclear fuel, it is necessary to use the widespread natural materials only. Existing technologies of thermal reactors lead to irretrievable spending of, mainly, zirconium and the attempts of wide use of breeders on the modern technological basis shall lead to increased irretrievable spending of such materials as chrome, nickel, manganese, tungsten. This fact is important for the future attempt of transition to use of thermonuclear reactors based on the electromagnetic confinement and also it might become substantial for the innovative technologies as well while assuming considerable increase in scale of nuclear power industry especially with regard to breeders with liquid-metal coolant.

Assuming that the number of reactors in the world increases by several-fold, transition to reprocessing of most of spent-fuel presents a problem in its own right.

Existing regulations covering discharges of alpha-active product by nuclear fuel cycle facilities will be revised that will prompt changes in relevant requirements.

Future increase in the thermal reactor fuel burn-up fraction due to an enhanced neutron economy may eventually help solve this problem. The increase of conversion factor in the reactor up to, for instance, the 0.85 (that is achievable in case of good neutron-physical characteristics in the thermal reactors) reduce the need of natural Uranium by two-to-threefold. This makes it possible to speak about partially closing the nuclear fuel cycle in the power reactor core.

In case of having a good neutron balance and the possibility to organize utilization of neutron leakage from the reactor core it would be possible to use a part of nuclear material for conversion of fissionable nuclides. In doing so, a part of nuclear material will not be a subject for enrichment or will not contain the (radiochemical) reprocessing products. To implement this idea the easily accessible free space available closely to and inside the reactor core is necessary and this should be combined with good neutron economy within the reactor core.

Such an approach requires increasing a number of fuel transpositions within and out of the reactor core and, hence, the nuclear reactor construction shall be adjusted for this. A part of fuel with high burn-up fraction will not need to be reprocessed.

4. INNOVATIVE PROJECTS REQUIRE TRUE NOVELTY

Use of well-known technologies with light and heavy water as well as of fast breeder reactors with liquid-metal coolant has got the limited capabilities and, probably, cannot resolve the contradiction, namely: to make a technology much safer and, at the same time, more economically competitiveness with the conventional power industry.

The solution to this problem could be found only by creating principally new technologies. In case of finding such technologies they will need to be supported so as to obtain evidence of their high performances within the restricted time.

It would, probably, be difficult to receive such support because for a long time key companies have invested their resources into the development of conventional technologies and, therefore, have a vested interest in pursuing traditional activities.

It is appropriate to recall that, practically, all currently available and well-developed technologies were produced while being supported by heavy government subsidies. There were military and later on – the civil programs.

It should be recognized that the really innovative technologies would require joint efforts of many countries.

5. SOLID COOLANT AS AN OPTION OF INNOVATIVE APPROACH

Study of information provided upon the Generation IV Program and the “INPRO” Requirements have led the RF Gosatomnadzor’ Management to the idea that it would be expedient to evaluate the still restricted investigation results obtained in the Tomsk-7 (now the Seversk, Russian Federation) collected starting from the end of 1980th and continued since 1994 at the “Luch” Enterprise (Podolsk, Moscow Region, Russia). The case in point is the materials justifying the use of solid coolant in nuclear reactors.

What makes this option attractive is the impression that the above dilemma could be resolved, to wit: to make a technology much safer, more environment-friendly and cheaper. One of the important features of solid coolant (if the material is able to withstand high temperatures as a solid one) is the absence of phase transient. This property makes the solid coolant rather simple and reliable from the point of view of prediction of its behavior in normal and emergency conditions. It also serves as a very good addition to the main properties of solid coolant, namely, to the possibility to have a pressure that is lower than the atmospheric one in conditions of very high temperature. The low pressure in combination with properties of solid coolant, in case of correct choice of constructive decisions, will mean the rather low probability of emergency processes and negligence of their consequences.

Materials with law capture cross-section that remain solid in the high temperature conditions do exist in the nature. The first but not only one of them is the graphite. It has a low capture cross-section, the very high sublimation temperature, up to a certain high temperature point its strength increases with temperature. Pyrolytic graphite does not support natural combustion up to temperatures above 1000 C. This is why it is widely used, in particular, in the missile technology. This element is widespread in nature. Pure graphite, practically, cannot be activated by the neutrons and has got negligible induced activity. Under such conditions it becomes clear why studies of solid coolant was started with graphite. Besides, the graphite has got a high thermal capacity, therefore the Reactor with thermal power of 3500 MW with an inlet coolant temperature of 450 C and outlet temperature of 850 C within the reactor core shall have the flow rate of 5 ton per second only.

At present the following might be considered as having been established with high degree of confidence:

- Optimal size and form of particles are defined;
- Conditions for organizing of reliable continuous movement of solid coolant via the Nuclear Reactor Core are formulated;
- The capability to organize continuous flow is experimentally proved;
- Flow velocity for continuous flow is defined;
- Density variations of solid coolant during its movement through the round channel are determined.

Of course, the values of acceptable flow velocity for solid coolant is considerably law than those for the gas and water coolant – it comes to dozens of centimeter per second.

Since available data related to flow velocity for solid coolant of chemical reactors, as a rule, are considerably law rather than those in nuclear reactors, and because of lack of data available in the literature with regard to heat transfer factor for the materials based on carbon selected as a coolant, for the flow velocity of 0.1 – 0.3 meter per second (that is acceptable for nuclear reactor), the necessity was recognized to build special installations and perform some experiments with the aim to obtain the values of heat transfer factors.

During the years of 1996-1999 the data related to heat transfer factors were obtained with acceptable estimated error. It was done for different types of gas medium in the experimental volume of argon, helium and vacuum.

Experimental studies of solid coolant wear under thermal cycling were started. By the end of 2002 the test duration reached 1000 hours during which the coolant was subjected to approximately 120 thousand cycles of passing through the heating part (that simulated the reactor core) and cooling zone (simulating the secondary heat exchanger).

These experiments demonstrated that based on the extent of coolant wear, one could predict its lifecycle as being on the order of 8-10 thousand hours.

Core neutronics analyses were performed for various solid coolant configurations. These calculations made it possible to select an acceptable graphite-to-uranium ratio in the core that meet the operational requirements and likely accident conditions.

6. WHAT DO WE GET BY USING SOLID COOLANT?

Since a reactor with this type of coolant is expected to operate in the high-temperature region for the purpose of utilizing high-temperature heat or for increasing its electrical efficiency and lowing the amount of waste heat, the pyrolytic carbon-coated particles are being considered now as fuel, similar to high-temperature gas-cooled reactor fuel.

During the experiments the temperature of heating wall was brought to 800 C and it is planned to increase after re-construction of installation.

The experimental values of the ultimate solid coolant continuous flow velocity, the heat-transfer factor as a function of the heating wall temperature and core neutronics analyses point out the feasibility to build the reactors with reactor core cooling by the solid coolant that is based on the carbon with thermal-power density up to 15 MW/m^3.

In principle, the reactor consists of the reactor core, the coolant hopper above the core, the heat exchangers allocated lower than the reactor core and the elevators/lifts that off-take the cooled solid coolant from below and drive it to the upper hopper. Thus, the coolant moves throughout the reactor core and heat exchangers under the gravity and returns back to the upper hopper by the elevator. The pressure of helium in the reactor core is a little bit lower that the atmospheric one.

Some reactor core components, the schemes of flow velocity regulation, the possibilities of reactor core emptying in case of shutdown of all lifting mechanisms as well as the adjustment of control rod drivers and reloading systems might be implemented in different manner and are the subject for future research and optimization.

Of course, while considering the design versions of reactor with the solid coolant the principal deficiencies are taken into account, namely;

— Lack of natural convection;
— Difference in dynamic and static density, possible variation of dynamic density in streamline of immovable components of reactor core and heat-exchanger;
— Deterioration and, as a consequence, the possibility of arising split or broken particles.

At the same time, the reactors with solid coolant might meet rather wide range of "User Requirements", in particular:

- Reduced probability of reactor component damage due to lack of overpressure in the reactor pressure vessel, low flow velocity of coolant and lack of considerable internal stress;
- Minor consequences of potential emergencies because of inherent safety of the Reactor Facility as well as sufficient sub-critical margin.
- Sufficient capabilities for studying and full-scope simulation of all operational and emergency regimes are available for such reactors;
- Reactor Facilities have a small metal capacity, good neutron balance, potential of free allocation of starting materials for obtaining new fissionable isotopes;
- No need to use the rare and expensive materials in the Reactor technology, such as chrome, nickel and zirconium;
- High heat economy additionally reduces the cost-per-unit and waste heat to the atmosphere;
- Use of high burn-up particle fuel, the possibility to use thorium due to availability of free space in the reactor core and nearby gives reason to consider this Project to be advanced from the viewpoint of nuclear non-proliferation.

7. INNOVATIVE PROJECTS WILL MAKE SAFETY MORE RELIABLE

The example of solid coolant potential use examination gives hope that another projects can be suggested as well. It is assumed that they would be of considerable difference with the existing one and, therefore, create a capability for long-term stable development of nuclear power technologies.

The matter concerns, first of all, the development of fuel elements based on another conceptual background as well as producing other concepts for the fast breeders. All the same, the stage of experimental proof is obligatory.

The proposed Guidelines INPRO Document, essentially, does not contain any attempts to establish more strong requirements with regard to Nuclear Installation Safety rather than the best from the existing one. This work should be done later on basing on proposals available. In this meaning the given example of solid coolant use reveals a precedent for making the safety regulations tougher. Such guidelines tightening is a logical consequence of two reasons:

- Desire of human society to live in more safe world;
- Increase the scale of the nuclear power industry. The growing up of its use shall not rise the real risk for all groups of population all over the world.

No matter how extensively nuclear power is used, its risk to the public should be infinitesimal. Effective innovative projects should help increase the scale of the nuclear power industry.

8. CONCLUSIONS

In our view the Final Draft Guidelines for the Evaluation of Innovative Nuclear Reactor and Fuel Cycles developed under the aegis of IAEA is an important and timely effort. User Requirements are formulated so that the nuclear power development problems, its role and capabilities are clear to the general public.

Regulatory agencies, being a part of user community, are responsible for forming of mandatory regulatory requirements for devices and processes.

The current stage of innovative technology selection requires extensive interaction between the regulators and engineers, designers, environmentalists, investors and politicians in order to formulate the basis for regulatory requirements imposed on innovative technologies and calls for international collaboration among various licensing authorities in order to formulate “Licensing Infrastructure” of their licensing activity and develop mutually acceptable underlying principles, approaches and, if possible, unified standards.

Representatives of developing countries where nuclear power will be developing at a faster than average pace will have to play an important role in this endeavor.

New regulatory standards should ensure that safety enhancements precede the increase in a scale of nuclear power development. It would be reasonable to develop such standards separately for individual innovative technologies on a long-term basis, for instance, for 15 years.

Releases and discharges into the environment especially releases of long-lived Alpha-sources require careful study. Given an increase in the scale of nuclear power utilization, such discharges may have a significant cumulative effect.

Preliminary analysis shows that the new technology - the use of solid coolant in nuclear reactors - could meet broad range of User Requirements for innovative technologies and, thus, serve as the basis for sustained development of nuclear power industry on an increasing scale.

It seems that development of innovative technologies requires a concerted effort of the world community.

ENVIRONMENTAL EFFECTS INCLUDING ALL THE STAGES OF THE FUEL CYCLE

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Abstract. Environmental impacts of all the stages of the nuclear fuel cycle were reviewed. Firstly, we reviewed the status of each fuel cycle stage from uranium mining to spent fuel disposal and the various endeavors to reduce the environmental impacts at the different stages of the nuclear fuel cycle. Secondly, the environmental friendliness of various fuel cycle options including some innovative fuel cycle concepts were investigated and compared. Lastly, we discussed some issues relating to the future nuclear fuel cycle developments.

1. INTRODUCTION

The Club of Rome predicted, in the famous report titled "The Limits to Growth" published in 1972, that the growth of the world economy would stop and reverse partly because of depleting nonrenewable energy resources[1]. However, technology development has extended the resource potential and it seems that resource limits will not be a constraint to growth, at least, for a few decades. A comprehensive report assessing energy and the sustainability issues was published by UNDP (United Nations Development Programme), UNDESA (United Nations Department of Economic and Social Affairs) and WEC (World Energy Council) in 2000 [2]. This report analyzed the social, economic, environmental and security issues linked to energy supply and use. One of the conclusions of the report was that physical resources were plentiful enough to supply the world's energy needs through the 21st century and beyond, but that their use may be constrained by environmental concern. Environmental impacts of energy supply and use become a crucial factor to sustainability of all kinds of energy forms including nuclear energy.

The most important impacts of any energy technology on the human environment are probably the impacts on health, safety and social well-being. In the case of nuclear power, public concern focuses on the effects of radiation to humans, which may be associated with some stages of the nuclear fuel cycle. Nuclear power emits virtually no pollution gases and generates much less amounts of waste than that of fossil fuel-based technologies. However, radioactive waste generated by nuclear power has caused public concern. The reliable and effective management of radioactive waste and spent fuel becomes a key factor for the continuous growth of the nuclear power program.

The main focus of this paper is on the environmental impacts of all the stages of the nuclear fuel cycle. Firstly, we reviewed briefly the status of each fuel cycle stage from uranium mining to spent fuel disposal and the various endeavors to reduce the environmental impacts at the different stages of the nuclear fuel cycle. Secondly, the environmental impacts of various fuel cycle options including some innovative fuel cycle concepts were investigated and compared. Lastly, some issues relating to future nuclear fuel cycle developments were discussed.

2. OVERVIEW OF THE ENVIRONMENTAL EFFECTS OF THE NUCLEAR FUEL CYCLE

2.1. Background

This section reviews briefly the current status of the nuclear fuel cycle with emphasis on its environmental aspects. All the stages of the fuel cycle can be divided into 2 categories such as the front-end fuel cycle and the back-end fuel cycle. Section 2.2 describes the front-end fuel cycle activities including uranium extraction, conversion and enrichment, and fuel fabrication. Section 2.3 describes the back-end fuel cycle activities such as spent fuel management, radioactive waste disposal and so on. MOX fuel fabrication is also considered as a back-end cycle activity for convenience.

2.2. Front-end fuel cycle

2.2.1. Uranium extraction

Uranium ore required to produce a ton of uranium depends on the average grade of the ore and typically amounts to about 10 to 1,000 tons ore/tU (grade 10%-0.1%U) [3]. The higher-grade deposits require a much lower rate of ore extraction but they require more cautious radiation protection measures for the workers because of their higher radiation exposure.

Mining of Uranium ore is commonly carried out by either underground or open pit techniques. Compared to underground mining, the amount of waste material is larger for the open-pit methods. A third method, the *in-situ* leaching (ISL) technology has a very low environmental impact because no ore is being brought to the surface during mining. However, its share is still limited to about 13% of the worldwide uranium production, because it requires some special conditions such as suitable sandstone-type deposits [4].

At the extraction stage of the cycle, the main environmental impact is mainly limited to the mining and mill tailings. The radiological impact is related mainly to the release of radon during mining and especially from the mill tailings; this impact accounts for a collective dose of 0.8-1.0 manSv per GWe-year [3]. So the main long-term environmental issue is the effective isolation of the daughter products of natural uranium, mainly radon decay products, from the environment.

2.2.2. Conversion and enrichment

Conversion plants are highly specialized chemical facilities and they handle some very aggressive chemicals (F, HF). They do not, however, produce significant amounts of radioactive effluents (principally containing natural uranium α-activity).

Very small quantities of uranium (U-234, 235, 238) are vented from the process and auxiliary systems of gaseous diffusion plants to the atmosphere while the radioactive discharges from centrifuge enrichment facilities are even smaller. For instance, the atmospheric releases from EURODIF in 1997 were 3.3 kg uranium, with a total activity of 0.16 GBq, and the liquid releases were only 0.29 kg uranium, with a total activity of 0.0094 GBq for uranium [4].

2.2.3. Fuel fabrication

Radon is generated from the natural decay of uranium. However, the conversion process removes all the uranium decay products, including radium, the direct parent of radon. Because the radium is removed and because the uranium and its daughter products that precede radon in the decay chain have a very long half-life, radium does not exist in the fuel and thus no radon is emitted which is the same for the enrichment process.

Only very small quantities of uranium are emitted from the fuel fabrication process to the environment. For instance, the atmospheric releases from the Romans UO2 fabrication plant in 1997 were only 0.0156 GBq, and the liquid releases were 2.644 GBq for uranium [4].

2.3. Back-end fuel cycle

The back-end of the fuel cycle starts when the fuel has been unloaded from the reactor and resides in the storage-ponds at the reactor site, waiting for spent fuel disposal or reprocessing. Because of the strategic and very long-term safety implications related with spent fuel management, the selection of an option for the back end of the fuel cycle is an important decision.

2.3.1. Interim storage and conditioning of spent fuel

Irradiated fuel assemblies are stored at reactor sites (AR) or away from reactors (AFR) at separate storage locations. Storage in water pools is the common practice for AR storage while AFR storage has been implemented in several countries as wet storage in pools or as dry storage using concrete canisters, metal casks or concrete vaults.

Practical experience from AFR wet storage pools shows that discharges of radioactive substances to the environment are very small and, thus, the radiological impacts of the discharges from AFR wet storage facilities to the public is negligible. Dry storage facilities for spent fuel assemblies show no or only very small discharges of radioactive substances to the environment [4].

The irradiated fuel assemblies that are not reprocessed have to be packed or conditioned prior to their disposal after a period of interim storage. The conditioning of spent fuel results in ILW and HLW. In general, 0.2 m^3 of ILW and 1.5 m^3 of HLW (i.e. the conditioned spent fuel) are created per ton of spent fuel.

2.3.2. Reprocessing and MOX fuel fabrication

To make use of the residual energy contents in the spent fuel, some countries have chosen to reprocess the spent fuel. Separation of plutonium from spent fuel could theoretically decrease the long-term radio-toxicity per unit of power by a factor of 10. However, recycling of plutonium as LWR-MOX only reduces the radio-toxicity of the assembly by a factor of 3, if the radio-toxicity of the spent LWR-MOX fuel is considered.

Current reprocessing practices do reduce considerably the total volume of waste for disposal. Per ton of spent fuel reprocessed, only 0.115 m^3 of the vitrified HLW waste and 0.35 m^3 of the ILW waste need to be sent to the disposal facilities. In addition, LLW waste undergoes super-compaction, leading to a much reduced volume of waste, which then can be included in the ILW waste for disposal. H^3 is the most dominant liquid waste, discharged from reprocessing plants in terms of activity and Kr^{60} is the most dominant gaseous waste [4].

A short-term issue of MOX fuel usage is that it impacts on the final waste management policy. The increased accumulation of minor actinides (especially Am^{241} and Cm^{244}) and the higher and longer-lasting residual heat in the fuel calls for an adapted waste management policy. In the longer term, multi-recycling of MOX fuel is a possible option, which is predicted not only to reduce the inventory of separated plutonium, but also to prevent an increasing fraction of minor actinides in the discharged fuel.

Gaseous and liquid releases from MOX fabrication plants are lower than the detection level, but it is estimated that they are in the order of 0.01 GBq/Gwe-year [4].

2.3.3. Radioactive waste disposal

Different waste forms arise from the nuclear fuel cycle. Those wastes fall into three categories: low, intermediate and high level waste. Low-level waste (LLW) comprises materials that do not have particularly long decay life nor do they produce a great deal of heat while decaying. Intermediate-level waste (ILW) includes mainly those arising in the fuel cycle facilities, containing medium active waste (Short-lived ILW), although some long-lived–emitters may be included (Long-lived ILW). High-level waste (HLW) comprises spent fuel or vitrified waste from reprocessing, including the long-lived and higher heat-emitting waste for geological disposal.

All wastes generated from the fuel cycle stages, except mining and milling, are sorted by contamination levels and conditioned at the site of its generation to facilitate subsequent handling. Radioactive wastes generated at the mining and milling stage are disposed of on the site.

LLW and short-lived ILW may be disposed of in near-surface repositories, while the preferred option for HLW disposal is a deep geological repository. In the management of spent fuel and radioactive waste, some important progress is worthy of note. In addition to the decision by Finland to construct a final disposal facility, the President of the USA approved the recommendation for a high level waste repository at Yucca Mountain, Nevada, and forwarded the recommendation to the US Congress for consideration.

In the last decade, significant progress has been made in the technical aspects of geological disposal. In general, the necessary technology for geologic disposal is available and can be deployed when public and political conditions will allow it. It is recognized, however, that there is relatively little experience in the application of some of these technologies and therefore demonstration and testing will continue and further refinements will be made. The technical progress has been facilitated through better integration of the main technical aspects of deep geologic disposal projects, namely the design of engineered systems, the characterization of potential disposal sites and the evaluation of total system performance.

Important advances have been achieved in the understanding of the performance of system components and their respective roles, the quantitative modeling of their behavior, treatment of uncertainty, the presentation of assessment findings, and the feedback of site selection, characterization and repository design. Much effort has been focused on increasing the reliability of the methods developed for confidence building/validation. Progress is also evident in the understanding of the natural system and in the characterization of potential sites. In particular, advances have been made in the measurement methodology and procedures, and in better appreciation for the heterogeneity (spatial variability) of the system.

In conclusion, the waste arising from the nuclear fuel cycle is well taken care of and technical solutions are either in place or under advanced consideration for implementation. While no specific problems occur for LLW and ILW waste disposal, public concern has delayed the implementation of HLW disposal projects.

2.3.4. Transport

Transport of nuclear materials, whether mined uranium or high-level waste, is a very important activity in the nuclear fuel cycle. All materials to be transported are packed in sealed containers. Non-irradiated material, such as Uranium concentrate and UO_2 are transported in drums, the fresh fuel elements in steel containers. For the chemically more hazardous material, UF_6 steel containers are used. Spent nuclear fuel from PWRs is mainly transported in special casks. Wastes are mainly packed in drums, placed into containers.

In order to evaluate the safety of the transport system, consideration is given to the risk of collisions, risk of fire, explosion and the combined action of fire and other failure modes, as well as the risk of immersion in water, and others. In France, for example, each year about 15,000 packages of radioactive materials, related to the nuclear fuel cycle are transported, of which only 750 contain fresh or spent fuel or HLW. Statistical data on nuclear transports in France during the period 1975-1997 indicate the occurrence, on average, of only one incident per year with some possible local impact, i.e. contamination of the transport container. Not one incident with radiological consequence has ever occurred in the OECD Member countries during the transport of fresh fuel, spent fuel or HLW.

3. ENVIRONMENTAL FRIENDLINESS IN ADVANCED NUCLEAR FUEL CYCLES

3.1. Backgrounds

Nuclear fuel cycle refers to the entire steps from the mining and milling of uranium ores, through to the manufacture of fuel elements, reactor operation, reprocessing of irradiated fuel and to the management of the wastes produced in all the stages of the cycle. From the entire fuel cycle, the nuclear power industry produces a number of environmental impacts similar to those which are caused by fossil fuels such as thermal pollution, land use, non-radioactive and radioactive emissions and occupational risks.

However, the main concern with releases from nuclear fuel cycle operations has been with radioactivity and its effect on the biosphere, and especially to human health. In this study, some indirect indicators for human health such as natural uranium requirements, spent fuel levels and radio-toxicity are dealt with in order to see how much the environment benefits from the advanced nuclear fuel cycles. Environmentally friendly nuclear fuel cycles being developed are divided into two groups considering their technological maturity. The first group is nuclear fuel cycles currently deployed or being deployed in the near future on an industrial scale. The second group is advanced nuclear fuel cycles with potential industrial deployment in more than 20 years, which may be called innovative nuclear fuel cycles. The environmental effects for the first and the second group are described separately in section 3.2 and section 3.3.

3.2. Nuclear fuel cycles deployed currently or in near future with industrial scale

The commercial nuclear fuel cycles in operation in the world include the once-through light water reactor (LWR) fuel cycle (e.g., U.S. and Sweden), once-through heavy water reactor (HWR) fuel cycle (e.g., Canada), the LWR fuel cycle with the recycled MOX (Mixed Oxide) fuel (e.g., Japan, France and Russia). The present civil use of recycled uranium and plutonium in LWR involves the development and utilization of large scale reprocessing plants and the MOX fuel fabrication facilities. There are some other alternative fuel cycles, which are currently used on a limited scale or are under development. They include the thorium fuel cycle (e.g., India), dry recycle and DUPIC (Direct Use of Spent PWR Fuel in CANDU) fuel cycle [5, 6]. In this section, we are going to introduce the recent KAERI's study [3] on the environmental effect of those fuel cycles. KAERI's study [7] aimed at examining whether the DUPIC fuel cycle will make radioactive waste management more effective and has environmentally friendly characteristics, compared with other conventional fuel cycles such as the PWR (Pressurized Water Reactor) once through cycle, CANDU (Canadian Deuterium Uranium) once-through cycle and thermal recycling option using existing PWR with MOX fuel.

3.2.1. Fuel cycle model and material flow

Figure 1 shows the fuel cycle options and steps or components consisting of the fuel cycles. The first cycle is low-enriched uranium in PWR of the once-through mode (hereafter called "PWR-OT"). The second cycle is mixed oxide fuel in PWR of the reprocessing mode (hereafter called "PWR-MOX"), in which spent PWR fuel is reprocessed and the recovered plutonium is used for making MOX fuel (5% of plutonium content) and the recovered uranium is used in a conversion plant. The MOX spent fuel will be disposed of without further plutonium or uranium recovery. Some depleted uranium generated in the enrichment plant will be used for making MOX fuel. The third cycle is the natural uranium in CANDU with the once-through mode (hereafter called "CANDU-OT"). The forth cycle is the DUPIC fuel cycle in which PWRs are linked to a CANDU (hereafter called "DUPIC"). The results of the material balance analyses extracted form the references [7] are also shown in Figure 1. All the values were expressed on the basis of 1 GWe-yr for all the fuel cycle options.

3.2.2. Natural uranium requirement and spent fuel levels

From the material flow of the Figure 1, we can compare natural uranium resources and spent fuel levels for each fuel cycle. The comparison result is shown in Figure 2. The relative amount of natural resources and spent fuel levels, as their maximum values, are also shown in the figure.

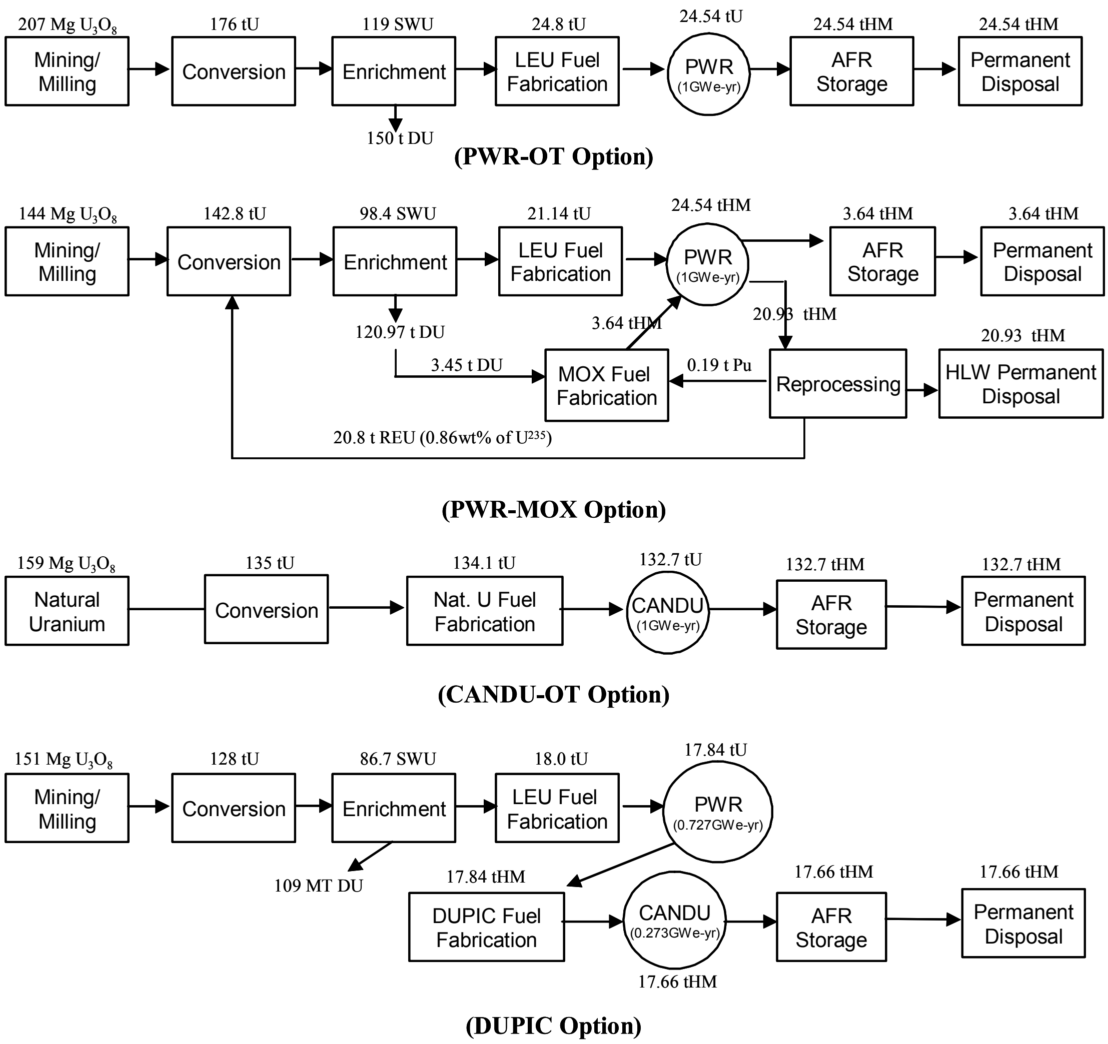


FIG. 1. Fuel cycle options and their material flows on the basis of one GWe-year.

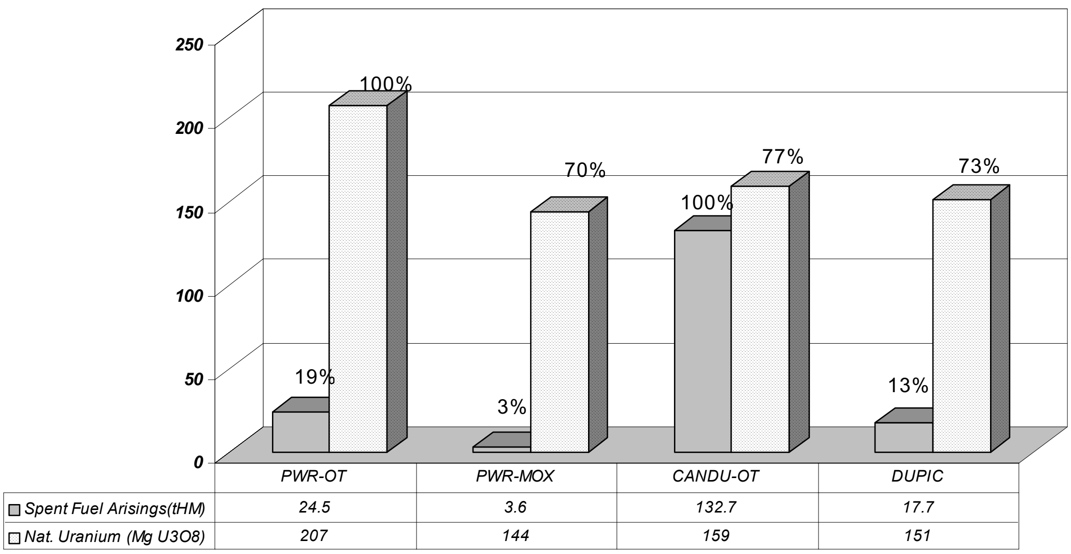


FIG. 2. Natural uranium resources requirement and spent fuel levels (1 GWe-year).

By comparison, it is shown that PWR-OT requires the largest natural uranium resources (207 Mg U_3O_8/GWe-yr) and CANDU-OT generates the largest spent fuels (~133 tHM/GWe-yr). It shows that the DUPIC fuel cycle with the PWR and CANDU reactor requires only 151 Mg U_3O_8 of natural uranium which is just for PWR fuel with an enrichment of 3.5 wt% ^{235}U. The PWR-MOX option has the smallest spent fuel level. In the PWR-MOX option, however, we must make sure that there is a high level waste to be disposed of. The DUPIC option has a ~27% uranium resource saving, compared with the PWR-OT fuel cycle. In addition, the amount of spent fuel annually discharged from the DUPIC fuel cycle generates only ~18 tHM/GWe-yr. It means that the DUPIC fuel cycle generates ~87% less spent fuel than that of the CANDU-OT cycle.

3.2.3. Total plutonium inventory

Total plutonium embedded in spent fuels can be calculated on the basis of 1 GWe-year as shown in Figure 3. Total plutonium inventory during 1 GWe-year is shown to be the biggest (~535 kg-Pu/GWe-yr) in the CANDU-OT option and the least (~88 kg-Pu/GWe-yr) in the PWR-MOX option. It means that the PWR-MOX option has some benefits in plutonium consumption. However, the DUPIC option contains ~141 kg-Pu/GWe-yr which is a little higher than the PWR-MOX case. On the whole, the CANDU-OT option has the largest fissile plutonium, which is a negative point for nuclear proliferation resistance.

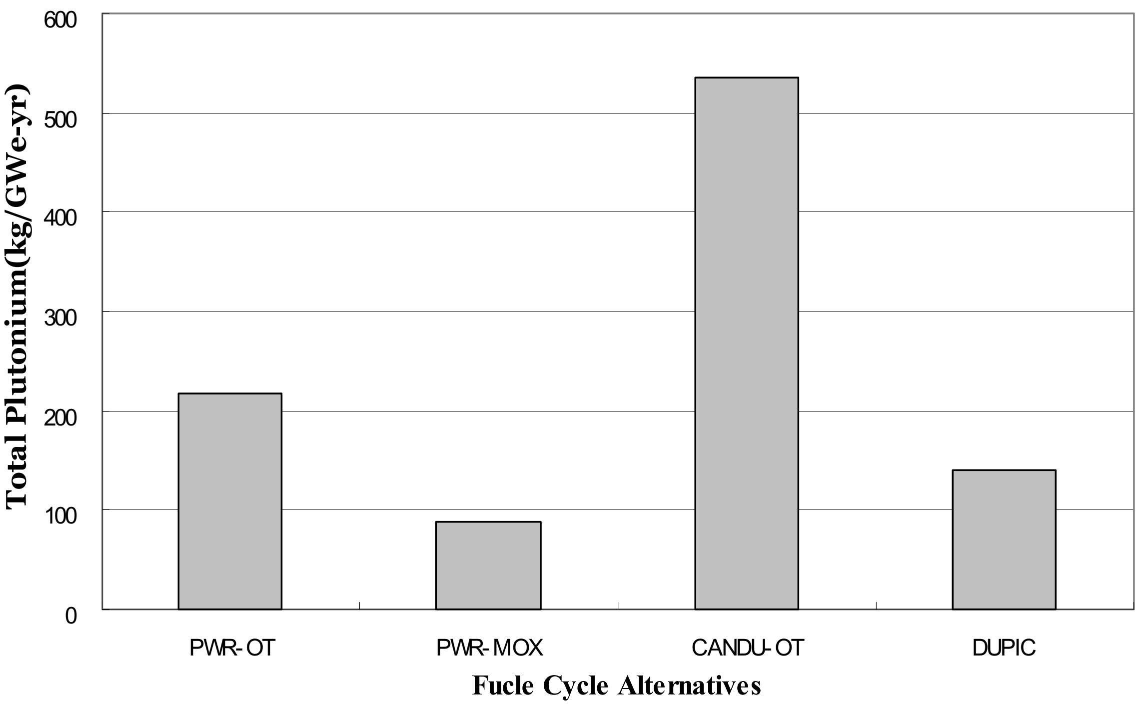


FIG. 3. Total plutonium inventory in various nuclear fuel cycles.

3.2.4. Radioactive toxicity

Figure 4 shows the comparison of the ingestion toxicity index for the fuel cycle options. The ingestion index is also compared with the toxicity of uranium ore mined for one GWe-year of reactor operation. From this figure, the ingestion toxicity of each fuel cycle option decays to a level below that of the initial ore after a period of about 1000 ~ 4000 years. It ultimately decays to a toxicity that is a fraction of a percent of the toxicity of the original ore consumed to generate these wastes. It indicates that the DUPIC option and the PWR-MOX option decay to a level below that of the initial ore after about 1000 years and 2000 years, respectively. We found that, up to that period, the toxicity of the DUPIC option is much smaller than other fuel cycle options.

In summary, we could say that the DUPIC fuel cycle could have good properties from the environmental effect aspect, compared to the other conventional fuel cycles, even though the differences are not so much.

3.3. *Advanced nuclear fuel cycles with potential industrial deployment in more than 20 years*

In the previous section, all the fuel cycles considered were kinds of conventional fuel cycles which are currently deployed or to be deployed in the near future on an industrial scale. This section will address advanced nuclear fuel cycles with potential industrial deployment in more than 20~30 years. These fuel cycles have been well described in the recent OECD/NEA report [3].

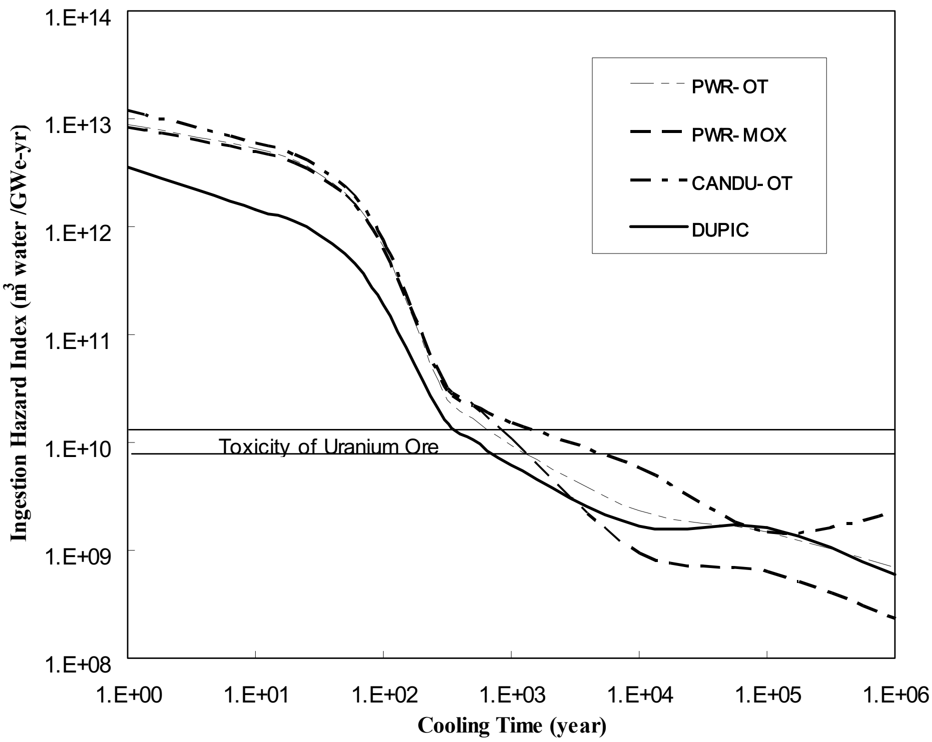


FIG. 4. Ingestion hazard index for various fuel cycle alternatives (1 GWe-yr).

3.3.1. *R&D on innovative nuclear fuel cycles*

Innovative nuclear fuel cycle alternatives are being studied for the future development of nuclear power in various countries including Korea, Japan, France and the U.S. In order to link those innovative fuel cycles to reactor technology, advanced reactor technology has also been developed. They include advanced light water reactors, high-temperature gas-cooled reactors employing the TRISO coated-particle type fuel system, fast reactors including KALIMER (Korea Advanced LIquid MEtal Reactor) and accelerator-driven systems including HYPER (HYbrid Power Extraction Reactor) in Korea. Of these advanced reactor systems, the ADS (Accelerator-Driven Systems) have emerged as promising concepts in terms of the environmental effect. That is because the accelerator-driven systems concepts are specifically designed for transmutation purposes in a partitioning and transmutation cycle.

The innovative nuclear fuel cycles considered recently are categorized into two groups from an environmental effect aspect. The first group consists of using fast reactor recycling not only for the plutonium but also for the minor actinides. The second group consists of concentrating the minor actinides into a separate part of the fuel cycle (second stratum) where these minor actinides are burnt in dedicated transmuters, i.e. ADS (Accelerator-Driven Systems). In this case, the plutonium would be burnt in fast reactors in the main fuel cycle (first stratum).

3.3.2. *Environment effect of innovative nuclear fuel cycles*

Most of the innovative fuel cycle concepts explicitly focus on the back-end and aim especially at dealing with the remaining waste. The main characteristics of the innovative fuel cycle concepts are to introduce additional waste management options, such as partitioning and transmutation, in order to reduce the mass and radioactivity of wastes going for final disposal. They are trying to close the fuel cycle not only for plutonium but also for the minor actinides.

In this section, the four potentially innovative fuel cycles described in the recent OECD/NEA report [3] will be compared in order to see how effective in terms of the environmental effect those nuclear fuel cycles are. We used the material flows of the report which are based on kg/TWhe. We converted the values to a new unit, kg/GWe-year, for consistency with the previous section. The four advanced fuel cycles are briefly described below.

3.3.2.1. Mixed LWR + FR(Fast Reactor) fuel cycle with homogeneous multiple recycling of TRU

In this cycle, fast reactors are designed as plutonium burners combining LWRs and FRs. The plutonium produced by irradiating the LWR fuels is reprocessed by using the chemical or pyroprocessing method and is burnt again in FRs. The fuels for FR are fabricated homogeneously with plutonium and depleted uranium. This option is hereafter called "LWR + FR (Homo)". It is assumed that LWR with a fuel burnup of 60 GWd/tHM and FR with a fuel burnup of 145 GWd/tHM have 56.1% park and 43.9% park, respectively, as described in the report [3].

3.3.2.2. Mixed LWR + FR fuel cycle with heterogeneous irradiation of MA targets

In this cycle, fast reactors are designed as plutonium and MA burners in a fuel cycle scheme combining LWRs and FRs. The actinides produced from LWRs are also reprocessed using the chemical or pyroprocessing method and are burnt in FRs. This option (hereafter called LWR + FR (Hetero)) has two different types of fuel, ie., MA (Minor Actinide) fuels and Pu/U fuels.

In this option, LWR with a fuel burnup of 60 GWd/tHM and FR with a fuel burnup of 145 GWd/tHM are assumed to have 44.3% park and 55.7% park, respectively.

3.3.2.3. 100% fast reactor park

This fuel cycle is to introduce a 100% fast reactor (hereafter called 100% FR). The plutonium is recycled indefinitely in the FR and the MAs are separated and sent for waste disposal. Such a fuel cycle scheme would allow the use of significant inventories of depleted uranium while minimizing the impact of the transuranics on the waste disposal performance. Adding a MA burner to this fuel cycle scheme would burn the long-lived minor actinides and reduce the impact even further. In addition, the FR fuel cycle is capable of recycling the TRU elements such as Pu and MAs as well as some fission products.

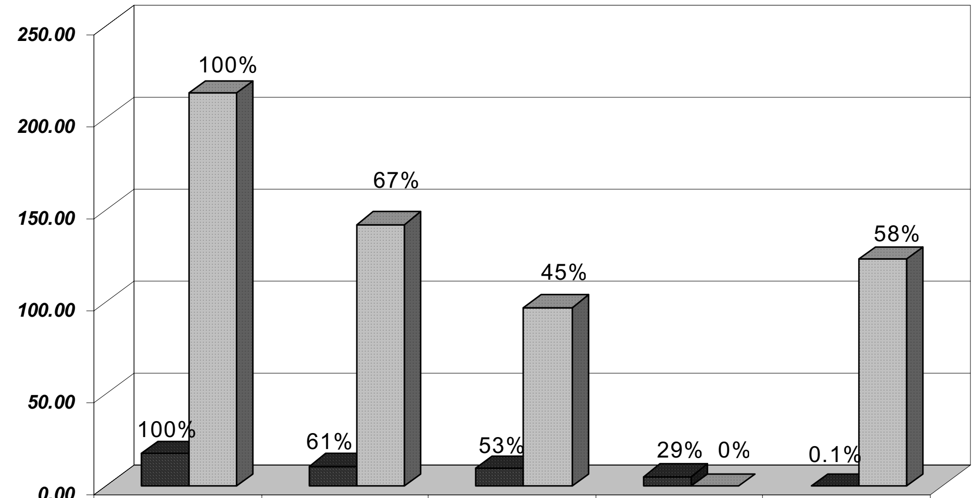
3.3.2.4. Double strata fuel cycle scheme with ADS

The so-called "Double Strata" fuel cycle scheme (i.e. OMEGA in Japan, and GEDEON in France) is an extension of the mixed LWR+FR fuel cycle scheme where all the separated minor actinides are treated in a second stratum and are transmuted by using an accelerator-driven system. This fuel cycle includes a fast reactor to burn the plutonium coming out of the UOX/MOX fuel cycle steps, with partitioning and transmutation of the minor actinides from the mixed LWR+FR fuel cycle which takes place in the second part of the cycle. The advantage of such a double strata scheme relates to the concentration of the hazardous highly radioactive radio-nuclides in a separate part of the fuel cycle where the dedicated reactor systems such as ADS can transmute them.

Figure 5 shows the comparison of uranium resources requirement and spent fuel level to be disposed of for above mentioned-innovative fuel cycles. Each fuel cycle is compared with the once-through option with 60 GWd/MTHM of burnup in order to see how effective each fuel cycle is. All the values (kg/TWhe) in the OECD/NEA report [3] are converted to kg/GWe-year so they could be compared with the results of the Figure 2.

Compared with the results of the conventional fuel cycle options in Figure 2, on the whole, the innovative fuel cycles have much more benefits in terms of natural uranium and spent fuel levels. It indicates that the LWR + FR(Homo), LWR + FR(Hetero), 100% FR and Double Strata options have 34%, 55%, 100%, 42% uranium resource savings, respectively, compared with the once-through option. The 100% FR option requires only depleted uranium for making nuclear fuels instead of natural uranium.

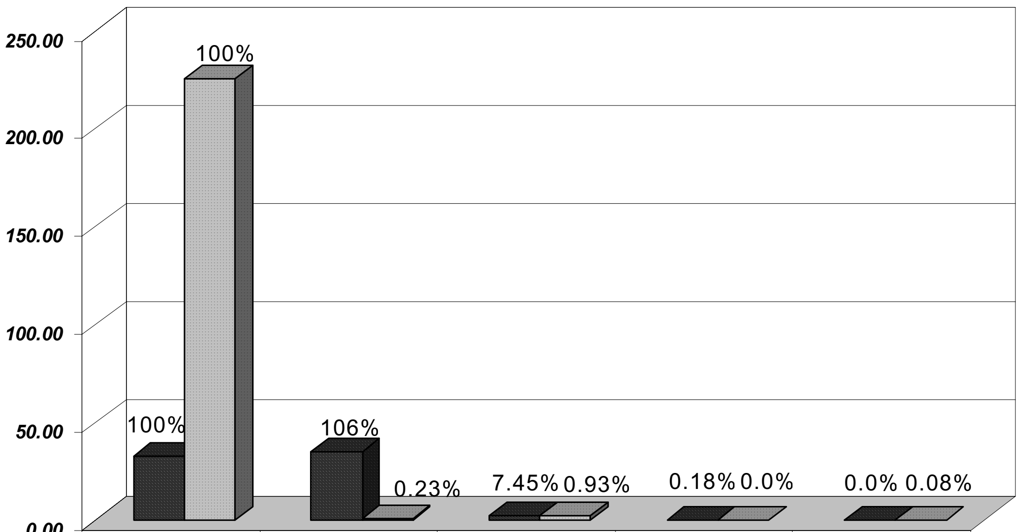
In the case of the spent fuel levels to be disposed of, it is found that spent fuels from the LWR + FR(Homo), LWR + FR(Hetero), 100% FR and Double Strata options could be reduced by 39%, 47%, 71% and 99%, respectively, compared with the once-through option. It indicates that the 100% FR option requires the smallest natural uranium resources while the Double Strata option generates the smallest spent fuels.



	Once through	LWR + FR (Homo)	LWR + FR (Hetero)	100% FR	double strata
Spent Fuel Arising (HTHM)	17.89	10.83	9.54	5.19	0.02
Nat. Uranium (Mg U3O8)	213.73	142.30	97.15	0.00	123.56

FIG. 5. Nat. Uranium resources and spent fuel levels for advanced fuel cycles.

Figure 6 shows the plutonium and minor actinide inventory based on 1 GWe-year which could be important elements for the long-term health impact in deep geological repository as well as long-term proliferation resistance. As shown in Figure. 6, plutonium generated during 1 GWe-year in the 100% FR cycle is shown to be completely consumed. However, MA generated in the Double strata option is also completely consumed. Other fuel cycle except for the LWR+FR (Homo) also have great benefits in terms of MA consumption.



	Once through	LWR + FR (Homo)	LWR + FR (Hetero)	100% FR	double strata
MA (kg/Gwe-year)	32.94	35.04	2.45	0.06	0.00
Pu (kg/Gwe-year)	225.57	0.53	2.10	0.00	0.18

FIG. 6. Plutonium and minor actinide inventory for advanced fuel cycles.

On the whole, we found from Figure 5 and 6 that the innovative fuel cycles have much more benefits in terms of the environment effect compared with the results of the conventional fuel cycle options in Figure 2.

4. CONCLUSION

Environmental impacts caused by nuclear power generation and its related fuel cycle activities become a key factor for future nuclear installation. Environmental concerns should be addressed for the sustainable development of nuclear power.

In this study, environmental impacts of all the stages of the nuclear fuel cycle were reviewed and some parameters on environmental impacts for conventional and advanced nuclear fuel cycles were investigated and compared.

Discharges from all the stages of the nuclear fuel cycle have been reduced in recent years due to the feedback from operating experiences and the application of new technologies.

It was found from the comparative study of environmental effects that the innovative fuel cycles have much more benefits than the conventional fuel cycle options.

Active R&D is underway on a number of new nuclear reactor and fuel cycle technologies and international cooperation would be helpful to facilitate those efforts.

5. REFERENCE

[1] MEADOWS, D.,H., MEADOWS, D.,L., RANDERS, J., BEHRENS III, W.,W., *The Limits to Growth*, Abstract established by Eduard Pestel. A Report to The Club of Rome (1972),

[2] UNDP, UNDESA and WEC, *World Energy Assessment: Energy and the Challenge of sustainability*, (2000)

[3] OECD/NEA, *Trends in the Nuclear Fuel Cycles; Economic, Environmental and Social Aspects*, (2001)

[4] OECD/NEA, *Radiological Impacts of Spent Nuclear Fuel Management Options; A Comparative Study*, (2000)

[5] CHOI, H.B., RHEE, B.W., PARK, H.S., *"Physics Study on Direct Use of Spent PWR Fuel in CANDU (DUPIC)*", Nucl. Sci. Eng.: 126, 80 (1997).

[6] WON IL KO, HANG-BOK CHOI, MYUNG SEUNG YANG, "*Economic Analysis on Direct of Spent Pressurized Water Reactor Fuel in CANDU Reactors – DUPIC Fuel Cycle Cost*", Nuclear Technology, Vol. 134, May (2001).

[7] WON IL KO, HO DONG KIM, MYUNG SEUNG YANG, "*Advantages of Irradiated DUPIC Fuels from an Environmental Impact Perspective*", Nuclear Technology, vol 138, No.2, 2002

ECONOMIC VIABILITY OF INNOVATIVE NUCLEAR REACTOR AND FUEL CYCLE TECHNOLOGIES*

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Abstract. Relative competitiveness of nuclear power, which varies depending on the regions and could be enhanced significantly by climate change regulations, is critically important to assess future potential growth of nuclear power. For a liberalized electricity market, average life-long power generation cost may not be the best criteria for utility investments. Short term return on investment (such as return on asset and cash flow) may become critically important. As a result, small and modular-type reactor could become preferred design for low growth and/or relatively small grid markets. But large reactor design can continue to be preferable for high growth and/or large grid market. Therefore it is important to assess market needs more carefully in determining the future advanced reactor design. International dialogue among various stakeholders is essential to understand better future requirements of next generation of nuclear reactors.

1. INTRODUCTION

Fifty years after the famous “Atoms for Peace” speech by US President Eisenhower in 1953, nuclear power has established its position as one of the most stable electricity supply sources in many countries in the world, supplying about 17% of total electricity generated. However, in order to keep that position, there are two important challenges that nuclear energy will face in the coming decades. They are; competition, and social/political confidence (including non-proliferation and terrorism). There is an increasing concern that existing nuclear technologies may not be able to overcome such tough challenges. It is expected that innovative technologies can be a part of the solutions to overcome such challenges. While we recognize that social and political conditions will be discussed by different speakers, this paper focuses on economic viability of innovative nuclear reactor and its associated fuel cycle technologies.

* Part of the work of this paper is based on the study sponsored by Agency of Natural Resources and Energy, Ministry of Economy, Trade and Industry, Japan [1].

2. GLOBAL LONG TERM ENERGY PATHS AND POTENTIAL ROLE OF NUCLEAR POWER

2.1. DNE global energy model up to 2050

While liberalization and competition are often the targets of nuclear industry strategy, it is important to consider the long term energy paths that may increase (or decrease) future potential role of nuclear power.

We applied the DNE21 global energy model, an optimization model consisting of 10 sub-regions, to four scenarios (A1, A2, B1, B2) developed by Intergovernmental Panel on Climate Change (IPCC) up to 2100 [2]. A1 scenario (global economy scenario) shows fastest energy demand growth, while B2 scenario (environment-emphasis, local economy scenario) shows the slowest energy demand. Both A2 scenario (population explosion scenario) and B1 scenario (environment-emphasis, global economy scenario) show a moderate energy demand growth. Among 10 regions, Asia (excluding Japan) has the highest energy demand growth in all scenarios. For comparative economic data of power generation, we used recent OECD/NEA report [3] except for Japan where government data is used [4]. Major assumptions for power generation costs, resource constraints etc. are summarized in Table I. We also conducted a sensitivity analysis for (a) with or without CO2 concentration constraints (at 550 ppm by 2100) and (b) nuclear power generation cost (increase/decrease).

Table I. Conditions for DNE21 simulation model

Duration of Simulation	2000 – 2100
Estimation of Resource and Cost	Coal: 6200Gtoe (55 - 310 $/toe) Oil: 810Gtoe (54 - 410 $/toe) N.G.: 840Gtoe (45 - 340 $/toe) Natural U: 15.4 Mt (40-130 $/kgU) Fossile fuel; estimation by Rogner (2000) Uranium; Uranium 1999 (OECD/NEA-IAEA)
Nuclear Fuel Cycle	Japan and Western Europe : Pu-recycle by Light Water Reactor is assumed Other Regions : Once-Through Cycle
Plant life	40 years

2.2. *Results of DNE model on nuclear power*

Regarding nuclear power, the model suggests the following.

— The potential growth of nuclear power can vary significantly, depending on the scenario, ranging from about two times from the current level, (under B2, without CO2 constraints) to 6 times (A1 with CO2 constraints), reaching to 12000 TWh in 2050 (from current 2600 TWh in 2000). The medium growth estimate is around 3~4 times (~7~8000 TWh) in 2050 (Figure 1).

— Highest growth is expected to occur in the Asian region (excluding Japan). Even under the most pessimistic scenario (B2), it could grow more than ten times (3000 TWh). For all cases, the share of Asian nuclear market is around 60-70% of total world nuclear power generation in 2050 (Figure 2). As to Japan, as well as for other developed countries, the growth estimate is much more modest. The highest growth estimate is about 3 times (A1 with CO2 constraints), and it could experience minus growth (B2 without CO2 constraints). In fact, most recent USDOE [5] and IEA forecasts [6] are very close to this scenario.

— In all cases, CO2 constraints as well as cost increase could change the growth estimate quite significantly. CO2 constraints (carbon tax on fossil fuels) could change relative competitive position of nuclear power significantly. For example, in Western Europe and in Asia, nuclear power will not be competitive without carbon tax, but will become competitive against both coal and natural gas after 2010 (Figures 3 and 4). Meanwhile, 20% cost reduction could increase global nuclear power growth by more double worldwide (Figure 5). In Japan, however, 20% cost increase could lead to nuclear phase out (Figure 6).

— Only in the high growth scenario, resource scarcity could become important factor after 2050, and thus fuel recycling and breeder reactor do not seem to play important role until then (Figure 7).

Given the above results, we found that economic competitiveness is certainly a critical factor to assess future role of nuclear power.

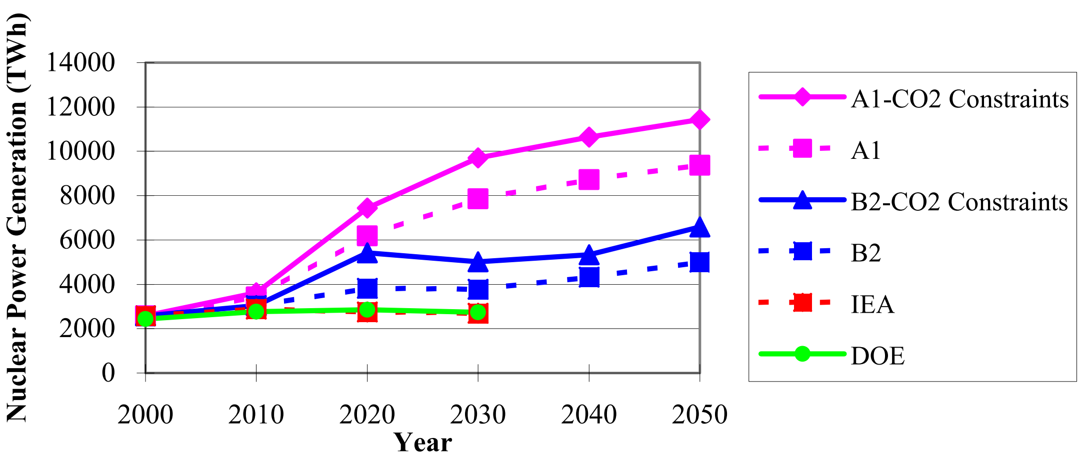


FIG. 1. Nuclear power generation prospects by DNE21 model.

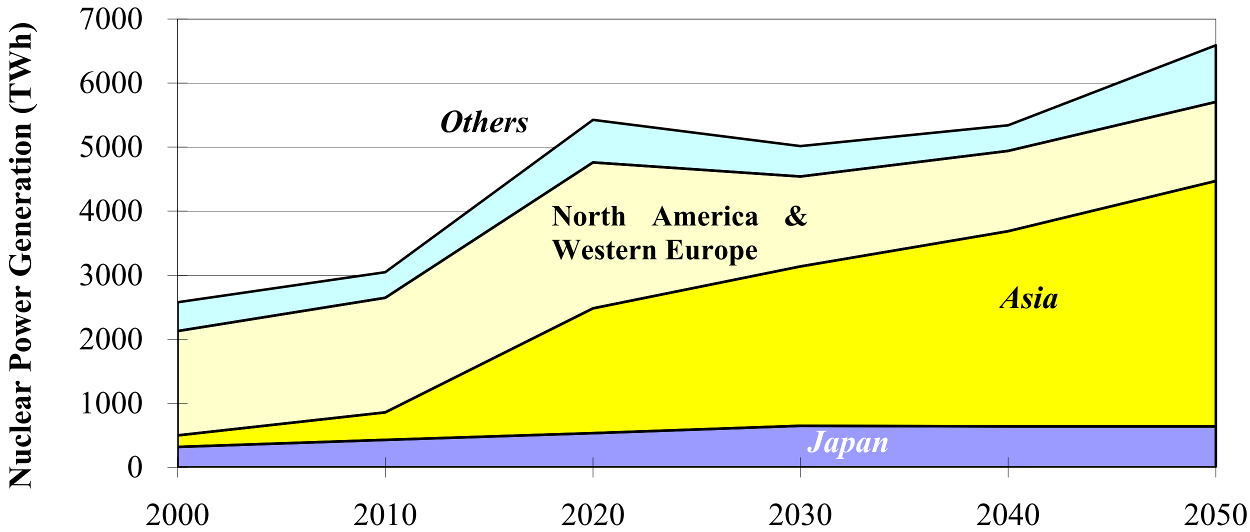


FIG. 2. Regional nuclear power generation prospects.

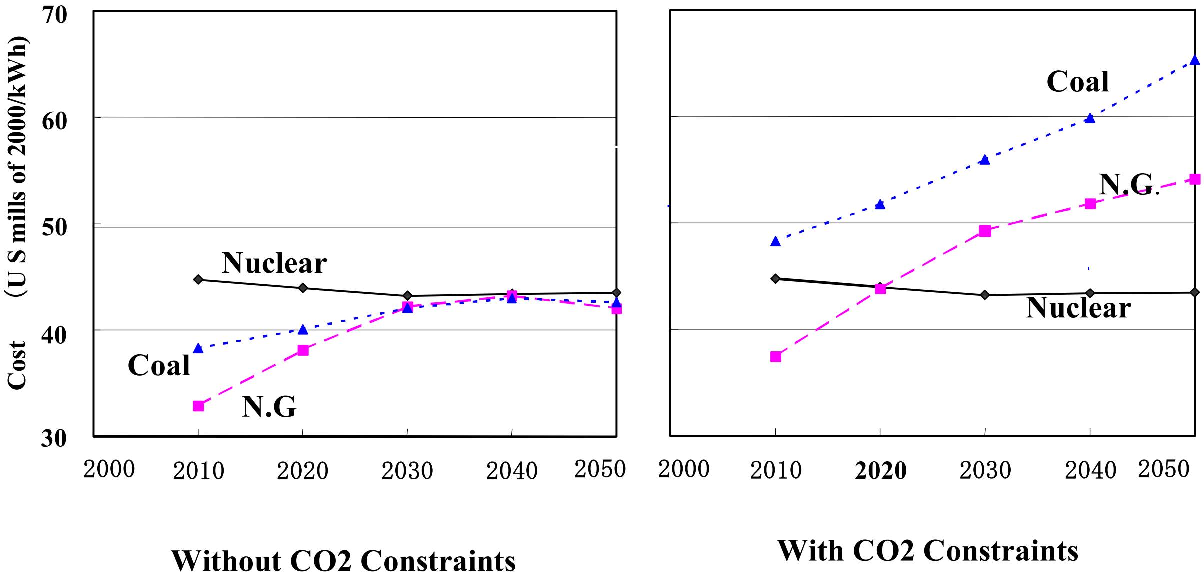


FIG. 3. Generation cost of each system in Western Europe.

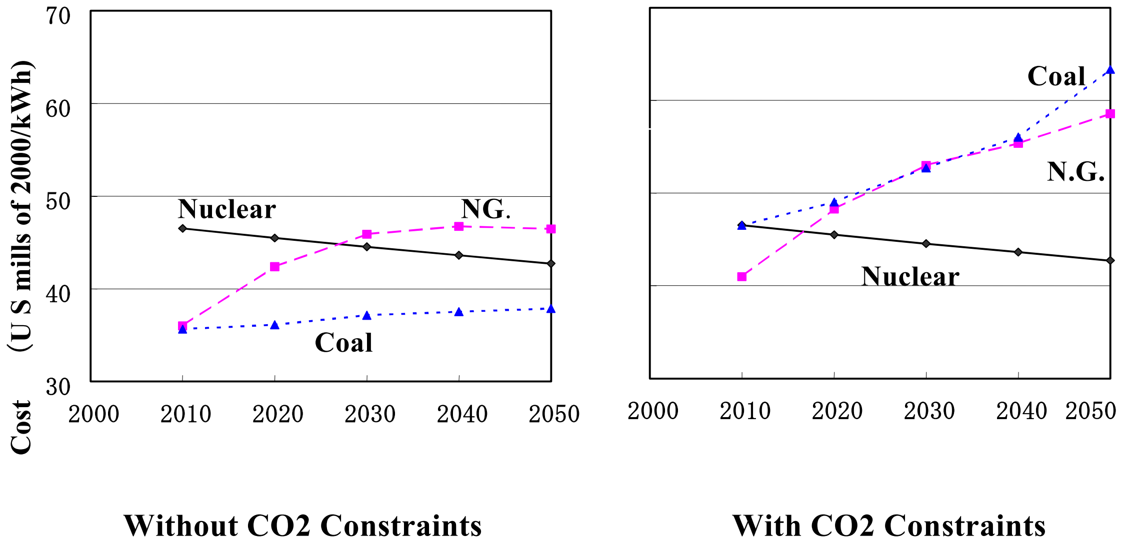
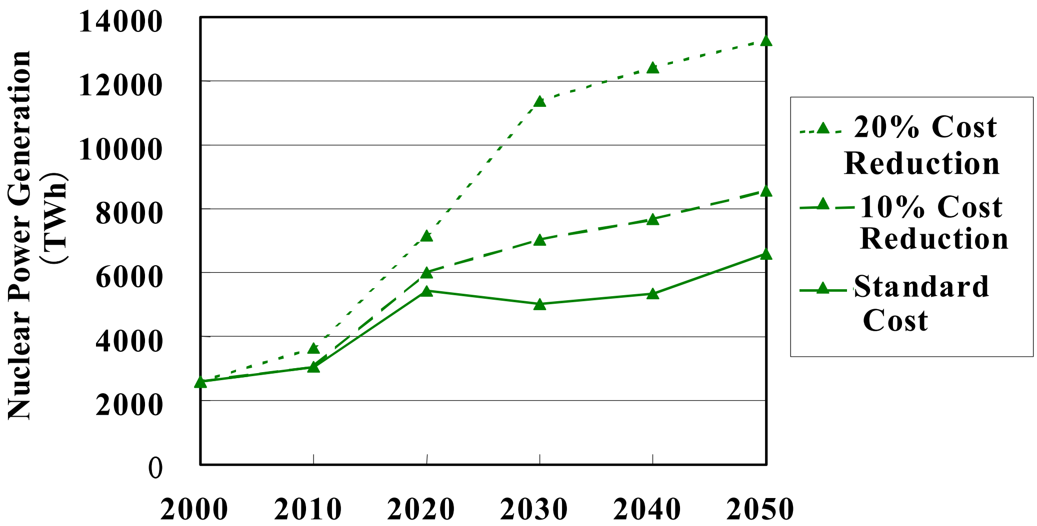


FIG. 4. Generation cost of each system in Asia.



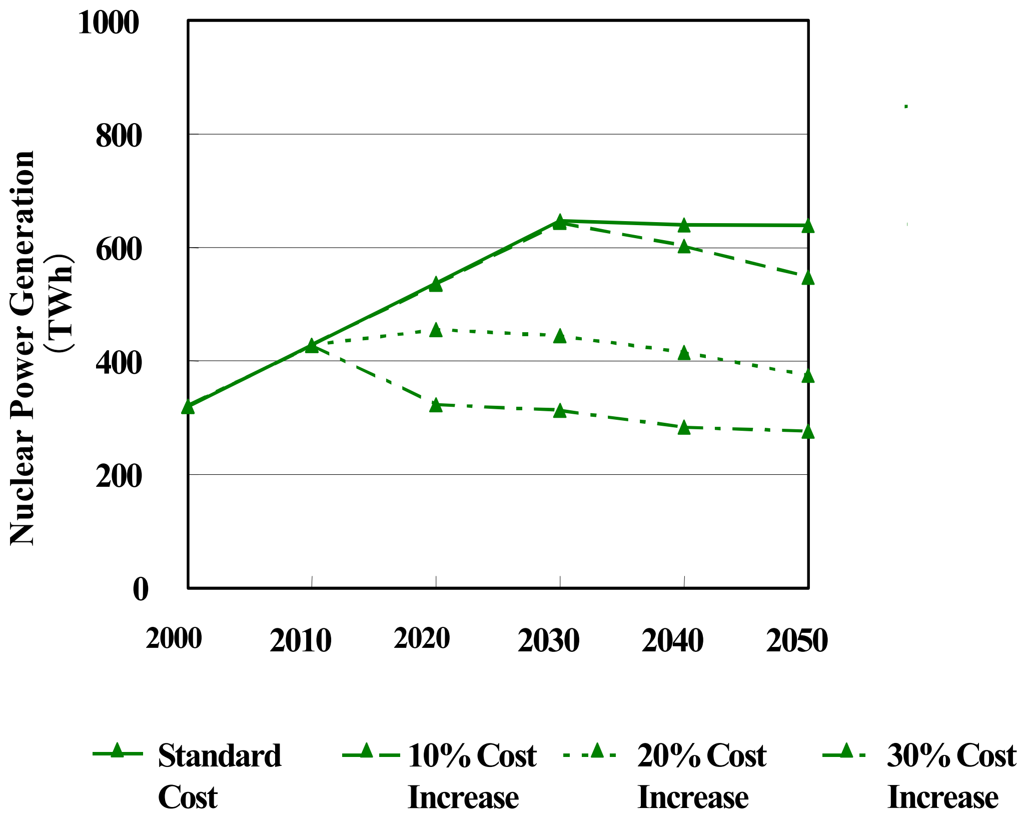


FIG. 6. The effect of cost increase on nuclear power generation prospects in Japan.

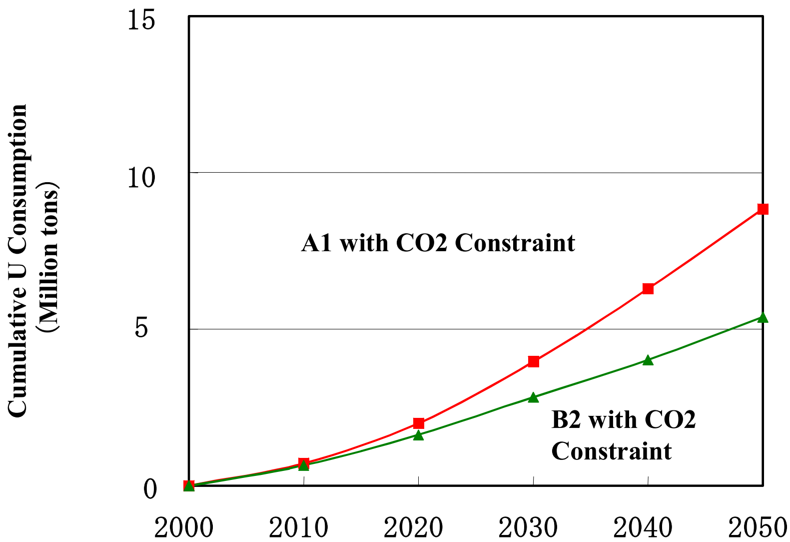


FIG. 7. Cumulative uranium consumption (No introduction of FBR is assumed).

3. ECONOMIC VIABILITY ASSESSMENT IN JAPAN

3.1. Japan's electricity market and nuclear power

In order to meet the potential growth shown by the above energy paths, however, nuclear power needs to be viable economically. In this section, we focus on Japan, where the market is being liberalized and future electricity growth is relatively mild as seen above. Energy security and climate change concern are important elements to consider Japan's nuclear power policy. In 2002, Japanese Diet passed the Energy Policy Basic Law, which clarifies the three main goals of Japanese energy policy, they are: supply stability, environmental preservation, and market mechanism, but priority is given to the first (supply stability) over the latter two goals [7].

Japan's electricity market has been partially liberalized since 1992, and its retail market will be completely liberalized after 2007. While vertical integration of existing power companies will be maintained, full liberalization of retail market will certainly make Japanese electricity market more competitive [8].

3.2. Investment criteria and requirements new reactor orders

Under the liberalized electricity market, life-long average generation cost ($/kWh) may not be the best criteria. Investment criteria for better profitability can become critically important for utility investors. For example, *return on asset (ROA)* within shorter time period could be more important investment criteria than life-long average generation cost since future market condition is much more uncertain than the regulated market. In Japan, ROA comparison based on the current data suggests that LNG power plant is more favorable than nuclear power, but it could change if construction cost is reduced or its capacity factor is improved significantly (Figure 8). As a result, in the liberalized electricity market, nuclear power will not be chosen as preferred power source, unless significant improvements are realized in capital cost and capacity factor.

Another important criteria under the liberalized market is "*cash flow*." Minimizing negative cash flow at any given time could be an important financial strategy. This suggests that smaller, modular-typed reactor design may be preferable for the competitive electricity market. This may be in particular true with low and/or uncertain growth market where modular type reactors could be more flexible in meeting uncertain power demand. While large reactor can still provide better return than small, modular-type reactor for growing and/or large grid market, small modular reactor can provide better cash flow if future demand is less than the expected demand since investment on small reactors can be adjusted to lower demand with less economic penalty.

Finally, one of the most uncertain factors for nuclear investment is back-end of fuel cycle. In Japan, reprocessing and decommissioning charges are still major uncertain cost factors as institutional arrangement on back-end of fuel cycle is still being debated. Therefore, uncertainty of back–end of fuel cycle cost is a critical factor assessing future of nuclear power in Japan. This may be also applied to other Asian market where back-end of fuel cycle cost is still uncertain.

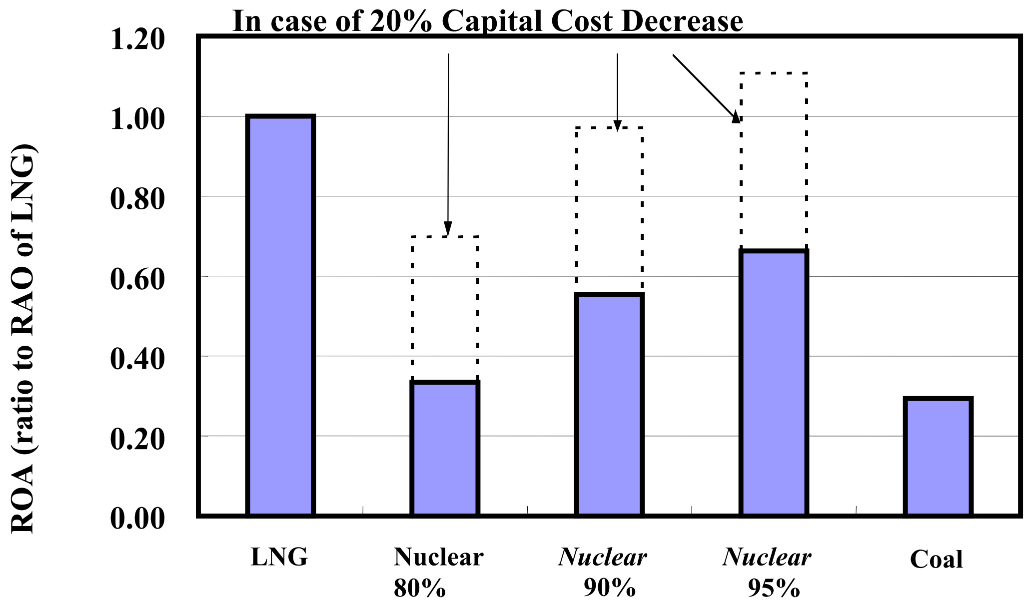


FIG. 8. ROA of each generation system during the first 5 years operation (discount rate: 3%).

4. IMPLICATIONS FOR ADVANCED REACTOR/FUEL CYCLE DEVELOPMENT STRATEGY

Given the results of above model analysis, following implications can be drawn.

— *CO2 constraints as well as power generation cost competitiveness could affect future growth of nuclear power quite significantly.*
Current trend suggests that nuclear power would not grow much without CO2 constraints, or even face minus growth if its power generation cost became higher. On the other hand, cost reduction with CO2 constraints could accelerate future expansion of nuclear power quite significantly.

— *In addition to life-long average generation cost, other investment criteria (such as return on asset and cash flow) may become critically important under the liberalized market.*
Under the liberalized electricity market, short term investment criteria could become more important than life-long average cost. This suggests that small initial investment is more acceptable than large capital investment. Advanced nuclear reactor need to incorporate such changes of electricity market.

— *This may suggest that small, modular-type reactor could be more advantageous than large scale, conventional reactor, especially in a low-growth and/or small grid market.*
Small initial investment leads to smaller reactor design if unit cost is the same as large unit. Modular type reactor means shorter construction time. Shorter refueling period is also advantageous since it could increase power productivity. This is especially true for low-growth and small grid market. A model cash flow analysis suggests that given the low (or uncertain) growth market, modular reactors have high economic advantage, while large scale reactor can enjoy scale-merit in faster growth market.

— *Given high growth and large grid market in Asia, large reactor design should not be excluded from advanced reactor designs.*

It is important to note that for fast-growing and/or large grid market large reactor may be more advantageous than small reactor. It is, therefore, very important to keep the large scale designs in advanced reactor programs.

— *Uncertainty in fuel cycle (back end) costs should be minimized.*
This is particularly important for Japan and for other Asian market where back end of fuel cycle program is not well developed. Institutional mechanism can help to reduce such uncertainty in fuel cycle costs, but reactor and fuel cycle design should also aim to minimize the uncertainty.

— *Breeding capability and/or fuel recycling criteria are not the highest priority at present, but could become important factor in high growth scenario, and after the latter half of century.*
Based on the global resource availability and growth potential of nuclear power, it can be concluded that breeding or recycling capability are not the highest priority at present for next generation of advanced nuclear reactor. On the contrary, it is important to assess economic viability of such designs very carefully under the different market conditions. Nevertheless, long term scenario suggests that breeding and recycling capability can become very important in the latter half of this century if nuclear power can enjoy the highest growth in the coming decades.

5. CONCLUSIONS: NEED FOR CLOSER DIALOGUE AMONG STAKEHOLDERS

Given the long term global energy scenarios, the future potential of nuclear power largely depends on regional energy demand growth, CO2 constraints, and relative competitiveness of nuclear power. In particular, carbon tax could change the relative competitiveness of nuclear power quite significantly and 20% reduction of nuclear power generation cost could also increase future nuclear growth substantially.

The analysis on Japanese market suggests that under the liberalized market new economic criteria, such as return on asset or cash flow, in addition to life-long average power generation cost, are necessary to assess future viability of nuclear power. The analysis also suggests that small, modular-type reactor may be more advantageous than large, conventional reactor in low-growth, liberalized electricity market. In Japan, uncertainty in back-end fuel cycle cost is also an important factor. However, in a fast growing and/or large-grid electricity market, large scale reactor may be more viable than small, modular-type reactor. Therefore, it is important to assess market needs more carefully in determining the future advanced reactor design.

In general, it is desirable to have a standardized reactor design all over the world, so that production scale merit can be maximized. However, it is also important to recognize that market condition and need may vary and thus criteria for reactor design may also vary.

Given the high risk of development of advanced reactor designs for future generation, therefore, it is critically important to keep close dialogue between users and reactor developers. We believe that current international efforts to develop future generation of reactors are noteworthy, but would like to call for much closer dialogue among other stakeholders, which could include regulators, NGOs and social scientists in addition to industry stakeholders. This will not only increase market acceptability of advanced reactor designs, but also increase future social acceptability of nuclear power.

6. REFERENCES

[1] THE INSTITUTE OF APPLIED ENERGY, "Investigation on next generation Light Water Reactor development strategy," (in Japanese), Tokyo (2003).

[2] INTERGOVERNMENTAL PANEL ON CLIMATE CHANGE (IPCC), Synthesis Report, Summary for Policymakers, http://www.ipcc.ch/pub/un/syreng/spm.pdf (2002).

[3] NUCLEAR ENERGY AGENCY/ORGANIZATION FOR ECONOMIC COOPERATION AND DEVELOPMENT, "Projected Cost of Generating Electricity: Update," Paris, (1998).

[4] ADVISORY COUNCIL ON ENERGY, NUCLEAR ENERGY SUBCOMMITTEE, Ministry of Economics, Trade and Industry (METI), "Cost Comparison of Power Plants," Tokyo (2000).

[5] US DEPARTMENT OF ENERGY/ENERGY INFORMATION ADMINISTRATION, "International Energy Outlook, 2003," (2003).

[6] INTERNATIONAL ENERGY AGENCY, "World Energy Outlook 2002," (2002).

[7] ENERGY POLICY BASIC LAW, Law No. 71, Japanese Diet, (2002).

[8] ISHIGURO, M., "Corporate Approaches to a Competitive Electricity Market," in M. Harris edit., "Energy Market Restructuring and the Environment: Governance and Public Goods in Globally Integrated Market," University Press of America, Maryland (2002)

PROLIFERATION RESISTANCE: POLITICAL FACTORS

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Abstract. Proliferation resistance measures, both intrinsic and extrinsic, could help ensure that future nuclear energy systems will continue to be an unattractive means to acquire materials for a nuclear-weapon programme, thus guaranteeing that *lack of trust does not result in technology-denial*. This paper describes the nuclear non-proliferation regime and the fundamentals of prevention of proliferation.

1. THE NUCLEAR NON-PROLIFERATION REGIME

The Treaty on the Non-Proliferation of Nuclear Weapons (NPT) recognizes rights and establishes obligations regarding the research, production and use of nuclear energy. On the one hand, all non-nuclear-weapon States parties to the NPT are required under the provisions of Article III.1 of the Treaty to bring into force a comprehensive safeguards agreement with the IAEA. On the other, Article IV guarantees the "inalienable right" of all the parties to the Treaty to develop applications of nuclear energy for peaceful purposes and in conformity with the non-proliferation obligations of the Treaty. This balance is the basis of the "deal" which has been accepted by the 189 States party to the NPT.

It is thus self-evident that the Agency's key functions under the technology pillar, as well as Article IV of the NPT, require *trust* in order to function optimally. This concerns for instance articles III.A.1 and III.A.2 of the Statute, which refer *inter alia* to the Agency's role in encouraging and assisting research, development and practical application of atomic energy for peaceful uses throughout the world.

Building the kind of trust that facilitates peaceful co-operation in the nuclear field is another of the Agency's fundamental roles. The Agency's safeguards system has always constituted an effective verification and confidence-building measure to provide assurances of non-diversion of safeguarded nuclear material. Moreover, as more and more States conclude and bring into force additional protocols, confidence in the effectiveness of the Agency's safeguards system is gradually being further strengthened, as the additional protocol provides the authority for verification activities to reach conclusions regarding the absence of undeclared nuclear material and facilities.

However, the lack of complete trust with respect to States' intentions regarding their nuclear programmes, and continuing advances in nuclear technology for enrichment and reprocessing, continue to be a major challenge for the nuclear non-proliferation regime.

For instance, one of the key principles of the Zangger Committee and the Nuclear Suppliers' Group, inter alia, is continuing vigilance. The members of the groups tend to operate on the side of caution, through the "non-proliferation principle"[1] which holds that notwithstanding other provisions of the guidelines, suppliers should authorize transfer of items or related technology identified in the trigger list *only* when they are satisfied that the transfers would not contribute to the proliferation of nuclear weapons or other nuclear explosive devices. In practice, this non-proliferation principle is generally interpreted as meaning that regardless of the type of recipient country, an export licence should be denied unless a supplier was satisfied that the transfer involved would not contribute to nuclear weapon proliferation.[2] This is sometimes referred to as a "presumption of denial", unless the requirements of the non-proliferation principle have been met fully.

While the IAEA is not a party to the NPT, it *is* a party to comprehensive safeguards agreements concluded by non-nuclear-weapon States pursuant to Article III.1 of the Treaty. The views of NPT States parties regarding safeguards, as expressed and agreed in the framework of the Treaty's review process, are communicated to the IAEA by the States parties through the Agency's policy making organs – the Board of Governors and the General Conference. Both in 1995 and in 2000,[3] NPT States parties affirmed that the IAEA is the competent authority responsible for verifying and assuring, in accordance with the Statute of the IAEA and the Agency's safeguards system, compliance with its safeguards agreements pursuant to Article III.1 of the NPT. NPT States parties noted their conviction that nothing should be done to undermine the authority of the IAEA in this regard. States parties that have concerns regarding non-compliance with the safeguards agreements of the treaty by the States parties should direct such concerns, along with supporting evidence and information, to the IAEA to consider, investigate, draw conclusions and decide on necessary actions in accordance with its mandate. The IAEA Director General has emphasized the importance of maintaining the authority, integrity and independence of multilateral verification organizations, as only they based on their verification and monitoring activities can provide credible assurances in accordance with their verification mandates. It is quite evident that national intelligence assessments regarding compliance do not have the authority or credibility of international verification organizations; particularly since the latter's conclusions are based on legally mandated verification and monitoring measures.

The development of truly proliferation-resistant nuclear energy systems could potentially make a major contribution to overcoming lack of trust, thereby contributing not only to the sustainable expansion of nuclear power and nuclear research, but also to the strengthening of the broader nuclear non-proliferation regime.

However, it is important that "proliferation resistant" technologies be developed in as transparent and inclusive a manner a possible, both in order to generate legitimacy and broader support, and to avoid competitiveness frictions.

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1 INFCIRC/254/Rev.2/Part 1 (October 1995), paragraph 11.
2 PPNN Briefing Book, Volume 1, Chapter 9, *Nuclear Export Controls*, pp. 56-57.
3 NPT/CONF.2000/28 (Parts I and II), (Article III, paragraph 7, page 3); http://disarmament.un.org/wmd/npt/1995dec2.htm, paragraph 9.

2. FUNDAMENTALS OF PROLIFERATION RESISTANCE

At the October 2002 Como meeting, agreement was reached on a number of definitions, fundamentals and principles in the area of proliferation resistance, which formed the basis for the report on that topic to the last INPRO Steering Committee. There was general understanding, inter alia, that:

— Proliferation resistance features and measures could help ensure that future nuclear energy systems will be an unattractive means to acquire materials for a nuclear-weapon programme;
— Proliferation resistance could be enhanced when complementary and redundant features and measures provide defence in depth;
— Proliferation resistance is likely to be most cost effective when an optimal combination of intrinsic features and extrinsic measures, compatible with other design considerations, can be included in a nuclear energy system; and
— Effective use of intrinsic proliferation resistance features would facilitate efficient application of extrinsic measures.

Proliferation resistance was defined as that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices. It was agreed that the degree of proliferation resistance results from a combination of, *inter alia*, technical design features, operational modalities, institutional arrangements and safeguards measures.

Intrinsic proliferation resistance features are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures.

Extrinsic proliferation resistance measures are those measures that result from States' decisions and undertakings related to nuclear energy systems.

The following extrinsic measures, inter alia, were identified:

— *States' commitments, obligations and policies with regard to nuclear non-proliferation and disarmament.*

These measures would include all relevant legal instruments, such as the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) and nuclear-weapon-free zone treaties. Although these treaties do not form an insurmountable barrier to proliferation, verification activities by the IAEA have overwhelmingly guaranteed compliance. In the case of the NPT, only once has a party made use of its right to withdraw from the Treaty. In addition, such legal commitments provide for continuity in the international non-proliferation regime by transcending government changes in States party to the Treaty.

Nevertheless, it should be noted that many of these measures work best as long as conditions remain static. History has shown that many of the non-proliferation policies of States and arrangements between States may change over time.

National export control legislation and co-operative arrangements, particularly those that limit nuclear energy use to peaceful purposes, constitute efficient extrinsic measures. The Zangger Committee, for instance, has developed common understandings concerning the interpretation and implementation of Article III.2 of the NPT, which regulates the provision of special material and equipment to States. In the same way, the Zangger Committee and the Nuclear Suppliers' Group have established, through their "trigger lists", export control principles designed to minimize the proliferation risk of nuclear exports.

— *Agreements between exporting and importing States to limit the use of nuclear energy systems to agreed purposes. This could be supported by an agreement between exporting and importing States that guarantees supplies of nuclear fuel or services.*

These measures include (1) bilateral arrangements for supply and return of nuclear fuel and other components of a nuclear energy system; (2) bilateral agreements governing the re-export of a nuclear energy system or its components by an importer; and (3) guarantees by a nuclear exporter of commercially attractive supplies of fresh fuel and waste management services over the life-cycle of the nuclear energy system, thus reducing the need of the importer to develop indigenous enrichment or reprocessing technologies.

Several countries have laws and regulations that limit the spread of sensitive knowledge or prevent the export of such knowledge and of sensitive equipment and materials in case certain conditions are not met. Many States do not export unless the recipient country has indeed accepted *full scope safeguards*.

— *Commercial, legal or institutional arrangements that control access to nuclear material and nuclear energy systems.*

These measures could include (1) the existence of a legal framework to ensure that operators of nuclear energy systems are subject to specific requirements governing the use of those systems and associated materials; (2) common legal provisions to be incorporated in all contracts involving nuclear energy systems; and (3) multi-national ownership, management or control of nuclear energy systems.

— *The application of IAEA verification and, as appropriate, regional, bilateral and national measures.*

These measures include the application of safeguards, for the detection – and deterrence – of diversion or undeclared production of nuclear material. The Agency's verification activities under the NPT are based on the comprehensive safeguards agreements that follow the INFCIRC/153 model agreement. Additional legal authority allowing the IAEA to implement further verification measures is conferred by the additional protocols to the safeguards agreements. The Agency's strengthened safeguards system has a confidence-building function that strongly contributes to proliferation resistance.

Naturally, for a verification system to be efficient and therefore credible, it requires adequate funding, technical competence and, as noted in the INPRO Report, an adequate number of sensitive and reliable measurement instruments and sensors.

— *Legal and institutional arrangements to address violations of nuclear non-proliferation or peaceful-use undertakings.*

These measures could include (1) a credible system of reporting verification conclusions in a timely manner; (2) reliable institutional arrangements for bringing evidence of violations before the international community; and (3) the existence of an effective international response mechanism.

3. CONCLUSION

The extrinsic measures mentioned above would be greatly complemented by intrinsic proliferation resistance features. Whereas the possibility of applying safeguards to and controlling exports of future nuclear energy systems will continue to play an important part – it is unlikely that we will have a fully proliferation-resistant system based only on extrinsic features. Thus, the development and implementation of intrinsic features should be encouraged.

Proliferation resistance measures, both intrinsic and extrinsic, could help ensure that future nuclear energy systems will continue to be an unattractive means to acquire materials for a nuclear-weapon programme, thus guaranteeing that *lack of trust does not result in technology-denial*. The benefits of enhancing proliferation resistance are not limited to the field of international security; by facilitating the access of developing States to nuclear technologies, proliferation resistance also could play a fundamental role in the field of development.

Finally, to ensure the widest possible acceptance and support for the concepts, principles and technologies for proliferation resistance, it is essential that "proliferation resistant" technologies be developed in as transparent and inclusive a manner, and as co-operatively, as possible.

PANEL ON CHALLENGES FOR THE DEPLOYMENT OF INNOVATIVE TECHNOLOGIES

(Session 4)

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WORLD NUCLEAR ASSOCIATION

(Results are included in Section 7 of the summary at the beginning of the publication

INTERNATIONAL PROGRAMMES
ON
INNOVATIVE NUCLEAR SYSTEMS

(Session 5)

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HISTORY AND STATUS OF THE GENERATION IV INTERNATIONAL FORUM

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Abstract. The Generation IV International Forum (GIF) has been active since January 2000. In its first three years, the GIF has worked with a flexible organizational structure and grown to ten member countries. The GIF has developed goals for Generation IV nuclear energy systems and completed a technology roadmap to define the most promising systems and their needed R&D. This paper describes the formation of the GIF and these accomplishments, and gives a current status on its activities to begin collaborative R&D on Generation IV systems.

1. FORMATION OF THE GIF

The first meeting of countries that eventually led to the formation of the GIF was held during January 2000 in Washington D.C. It was attended by nine countries, and began with a general discussion of their common interest in new nuclear energy systems. The meeting resulted in a joint press release of the need for a next generation, known as Generation IV, and their desire to cooperate in research and development.

The figure below gives an overview of the generations of nuclear energy systems. The first generation was advanced in the 1950s and 60s in the early prototype reactors. The second generation began in the 1970s in the large commercial power plants that are still operating today. Generation III was developed more recently in the 1990s with a number of evolutionary designs that offer significant advances in safety and economics, and a number have been built, primarily in East Asia. Advances to Generation III are underway, resulting in several (so-called Generation III+) near-term deployable plants that are actively under development and are being considered for deployment in several countries. New plants built in the nearer term will likely be chosen from these plants. In the longer term, the prospect for innovative advances through renewed R&D has stimulated interest worldwide in a fourth generation of nuclear energy systems.

The ten countries joined together to develop future-generation nuclear energy systems that can be licensed, constructed, and operated in a manner that will provide competitively priced and reliable energy products while satisfactorily addressing nuclear safety, waste, proliferation, and public perception concerns. The objective for Generation IV nuclear energy systems is to have them available for international deployment by the year 2030 or earlier.

From the beginning, the desire of the member countries was to have a fairly flexible arrangement for working together to define Generation IV systems and their R&D. The basic elements for a GIF Charter were discussed at subsequent meeting in August 2000 and March 2001. By July 2001, representatives of Argentina, Brazil, Canada, France, Japan, the

Republic of Korea, the Republic of South Africa, the United Kingdom, and the United States signed a Charter.[1] The GIF operates with a Policy Group that acts as a decision-making body for high-level initiatives and issues, and an Experts Group that acts as oversight to the R&D collaborations. Both groups operate with representatives from each member country. The Charter has provisions for a modest Secretariat initially hosted by the United States. There are no permanent facilities or staff, and the member countries contribute part time staff to attend meetings and develop the documents that the GIF has produced. The Charter contains steps for the addition of new members, and in February 2002, Switzerland was welcomed into the GIF.

The primary activities of the GIF are to:

- Identify potential areas of multilateral collaborations on Generation IV nuclear energy systems,
- Foster collaborative R&D projects,
- Establish guidelines for the collaborations and reporting of their results,
- Regularly review the progress and make recommendations on the direction of collaborative R&D projects,
- Establish and regularly review an inventory of the potential areas of needed research, and
- Conduct such other activities to advance achievement of the GIF's objective as the members may jointly determine.

From the early interactions of the members, it was apparent that the objectives of the R&D deserved further development. Two activities were spawned: the Policy and Experts Groups began working on a set of goals for Generation IV systems, and a major activity was started to produce a technology roadmap. Roadmapping is a methodology used to define and manage the planning and execution of large-scale R&D efforts.

The GIF agreed to support the preparation of a roadmap, and the roadmap became the focal point of their efforts. The organization and execution of the roadmap became the responsibility of a Roadmap Integration Team. More than one hundred technical experts from the ten member countries contributed to its preparation. The scope of the R&D described in this roadmap covers all of the Generation IV systems. However, each GIF country will focus on those systems and the subset of R&D activities that are of greatest interest to them. Thus, the roadmap provides a foundation for formulating national and international program plans on which the GIF countries will collaborate to advance Generation IV systems.

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[1] The GIF Charter is available at http://gen-iv.ne.doe.gov/pdf/GIF-CHARTER-7-5-01-revised.pdf

2. GOALS FOR GENERATION IV

As preparations for the Generation IV Technology Roadmap began, it was necessary to establish technology goals for these nuclear energy systems. The goals have three purposes: First, they serve as the basis for developing criteria to assess and compare the systems in the technology roadmap. Second, they are challenging and stimulate the search for innovative nuclear energy systems—both fuel cycles and reactor technologies. Third, they will serve to motivate and guide the R&D on Generation IV systems as collaborative efforts get underway.

Eight goals for Generation IV were defined in the four broad areas of sustainability, economics, safety and reliability, and proliferation resistance and physical protection. Sustainability goals focus on fuel utilization and waste management. Economics goals focus on competitive life cycle and energy production costs and financial risk. Safety and reliability goals focus on safe and reliable operation, improved accident management and minimization of consequences, investment protection, and essentially eliminating the technical need for off-site emergency response. The proliferation resistance and physical protection goal focuses on controlling and securing nuclear material and nuclear facilities.

The eight technology goals[2] for Generation IV nuclear energy systems are given below:

— *Sustainability–1.* Generation IV nuclear energy systems including fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.
— *Sustainability–2.* Generation IV nuclear energy systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.
— *Safety and Reliability –1.* Generation IV nuclear energy systems operations will excel in safety and reliability.
— *Safety and Reliability–2.* Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
— *Safety and Reliability–3.* Generation IV nuclear energy systems will eliminate the need for offsite emergency response.
— *Economics–1.* Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
— *Economics–2.* Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
— *Proliferation Resistance and Physical Protection–1.* Generation IV nuclear energy systems including fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials and provide increased physical protection against acts of terrorism.

A central feature of the roadmap is that the eight goals of Generation IV are all equally important. That is, a promising concept should ideally advance each, and not create a weakness in one goal to gain strength in another. On the other hand, promising concepts will usually advance one or more of the goals or goal areas more than others.

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[2] The full technology goals document is available at http://gen-iv.ne.doe.gov/pdf/finalgenivgoals_may01.pdf

3. THE GENERATION IV ROADMAP PROJECT

As the Generation IV goals were being finalized, preparations were made to develop the Generation IV technology roadmap. The organization of the roadmap is shown in the Figure 1. The Roadmap Integration Team (RIT) was the executive group. Groups of international experts were organized to undertake identification and evaluation of candidate systems, and to define R&D to support them.

In a first step, an Evaluation Methodology Group was formed to develop a process to systematically evaluate the potential of proposed Generation IV nuclear energy systems to meet the Generation IV goals. At the same time, a solicitation was issued worldwide, requesting that concept proponents submit information on nuclear energy systems that they believe could meet some or all of the Generation IV goals. Nearly 100 concepts and ideas were received from researchers in a dozen countries.

Technical Working Groups (TWGs) were formed—covering nuclear energy systems employing water-cooled, gas-cooled, liquid-metal-cooled, and nonclassical reactor concepts—to review the proposed systems and evaluate their potential using the tools developed by the Evaluation Methodology Group. Because of the large number of system concepts submitted, the TWGs collected their concepts into sets of concepts with similar attributes. The TWGs conducted an initial screening, termed *screening for potential*, to eliminate those concepts or concept sets that did not have reasonable potential for advancing the goals, or were too distant or technically infeasible. Following the screening for potential, the TWGs conducted a *final screening* to assess more quantitatively the potential of each concept or concept set to meet the Generation IV goals.

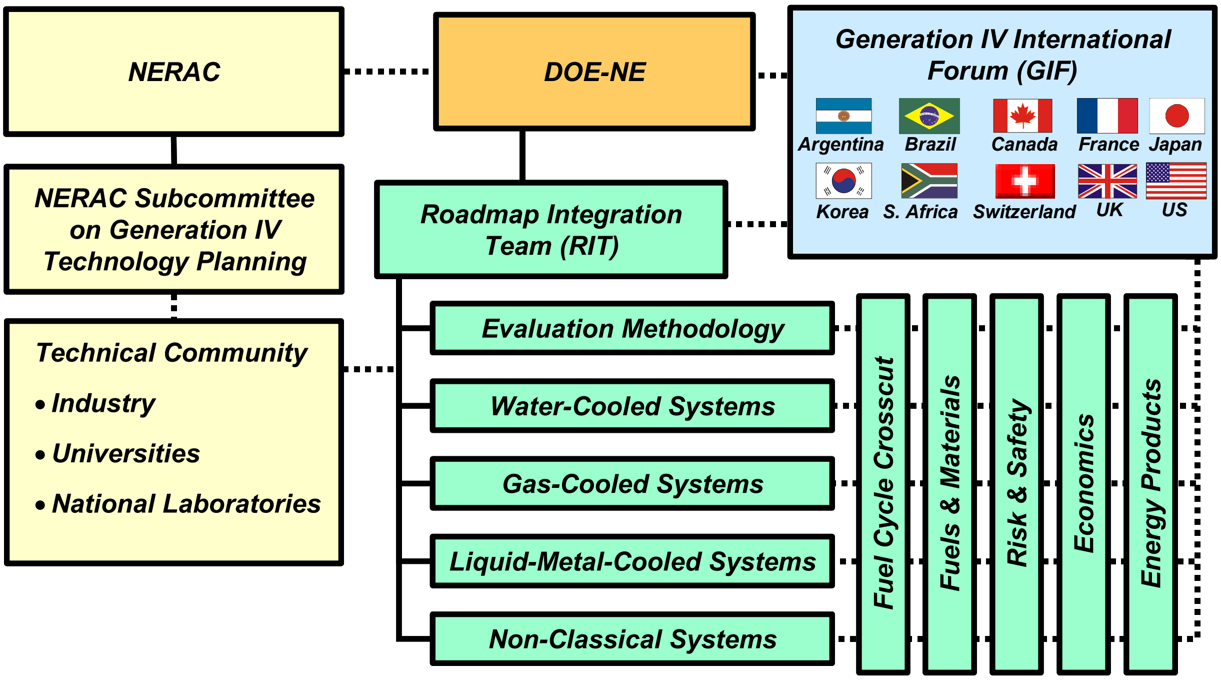

FIG. 1. Organization of the rodemap.

A Fuel Cycle Crosscut Group (FCCG) was also formed at a very early stage to explore the impact of the choice of fuel cycle on major elements of sustainability—especially waste management and fuel utilization. Their members were equally drawn from the working groups, allowing them to compare their insights and findings directly. Later, other Crosscut Groups were formed covering economics, risk and safety, fuels and materials, and energy products. The Crosscut Groups reviewed the TWG reports for consistency in the technical evaluations and subject treatment, and continued to make recommendations regarding the scope and priority for crosscutting R&D in their subject areas. Finally, the TWGs and Crosscut Groups worked together to report on the R&D needs and priorities of the most promising concepts.

The international experts that contributed to this roadmap represented all ten GIF countries, with several experts from the Organisation for Economic Cooperation and Development Nuclear Energy Agency, the European Commission, and the International Atomic Energy Agency.

The selection of the systems to be developed as Generation IV was accomplished in the following steps:

1. Definition and evaluation of candidate systems
2. Review of evaluations and discussion of desired missions (national priorities) for the systems
3. Final review of evaluations and performance to missions
4. Final decision on selections to Generation IV and identification of near-term deployable designs.

The use of a common evaluation methodology is a central feature of the roadmap project, providing a consistent basis for evaluating the potential of many concepts to meet the Generation IV goals. The Evaluation Methodology Group developed the methodology at an early stage in the project.[3] The basic approach was to formulate a number of factors that indicate performance relative to the goals, called *criteria*, and then to evaluate concept performance against these criteria using specific measures, called *metrics*.

Near the end of the first step, the GIF met to conduct the second step of the selection process in February 2002. The GIF reviewed the preliminary evaluation results and discussed additional considerations that would be important to their final decision. These included a review of the important conclusions of the fuel cycle studies, which helped to suggest the various missions for Generation IV systems that were of interest: electricity and hydrogen production and actinide management.

A final review of the evaluations and performance to missions by the GIF Experts Group completed the third step in April 2002. The GIF met in May and July 2002 to conduct the fourth step. In brief, the candidate concepts that emerged from the final screening were discussed. Each was introduced with a presentation of the concept in terms of final evaluations, performance of missions, and estimated deployment dates and R&D costs. The Policy members discussed the concepts until a consensus was reached on six systems found to be the most promising and worthy of collaborative development.

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[3] The Evaluation Methodology is available at:
http://gif.inel.gov/roadmap/pdfs/012_final_screening_evaluation_methodology_report.pdf

4. GENERATION IV NUCLEAR ENERGY SYSTEMS

The Generation IV roadmap process culminated in the selection of six Generation IV systems.[4] The motivation for the selection of six systems is to:

— Identify systems that make significant advances toward the technology goals
— Ensure that the important missions of electricity generation, hydrogen and process heat production, and actinide management may be adequately addressed by Generation IV systems
— Provide some overlapping coverage of capabilities, because not all of the systems may ultimately be viable or attain their performance objectives and attract commercial deployment
— Accommodate the range of national priorities and interests of the GIF countries.

Six systems were selected to Generation IV by the GIF (Table I).

5. PREPARATIONS FOR FUTURE R&D

Now that the Generation IV Technology Roadmap has been completed, the GIF is currently planning for collaborative R&D projects. Steering committees have already been formed for most of the systems that will guide the R&D and oversee collaborative efforts. Crosscutting research needs are being considered by the Experts Group, along with a Management Board that will oversee collaborative efforts.

Anticipating the need for more a formal structure as collaborative R&D projects become active, the GIF has created a Task Force on Multilateral Agreements to consider governing agreements that will become the basis for collaborations. While a few bilateral R&D collaborations are already underway, others are expected to follow within the next year under existing agreements. It is expected that formal agreements may take some time to complete.

Table I. Selected system in alphabetically order.

Generation IV System	**Acronym**
Gas-Cooled Fast Reactor System	GFR
Lead-Cooled Fast Reactor System	LFR
Molten Salt Reactor System	MSR
Sodium-Cooled Fast Reactor System	SFR
Supercritical-Water-Cooled Reactor System	SCWR
Very-High-Temperature Reactor System	VHTR

[4] The Generation IV Technology Roadmap is available at:
http://nuclear.gov/geniv/Generation_IV_Roadmap_1-31-03.pdf

6. SUMMARY

In summary, the GIF has developed from an initial discussion between countries with a common interest in new nuclear energy systems into a developing framework for collaborations on the R&D needed to make definite advances toward Generation IV systems. The framework now includes a set of technology goals, a technology roadmap for Generation IV systems, and the first steps toward collaborative R&D programs.

BACKGROUND AND STRUCTURE OF THE INTERNATIONAL PROJECT ON INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES (INPRO)

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Abstract. In response to an IAEA General Conference Resolution in September 2000, which invited both nuclear technology suppliers and users to combine their efforts to consider international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles, the IAEA launched the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) in May 2001. The General Conference of the IAEA in September 2001 praised the initial progress of INPRO and adopted another resolution, which recognized the unique role that the IAEA can play in international collaboration in the nuclear field. As of June 2003, 14 IAEA Member States and the European Commission have become member of INPRO. In total, more than 20 cost-free experts have been nominated by these Member States and the European Commission to work for the INPRO project at the IAEA. Five meetings of the INPRO Steering Committee (SC), which is the decision making and review body of INPRO, were held. The 5th SC meeting was held on 26-28 May 2003 to review results and to recommend further actions. The objective of INPRO, which comprises two phases, is to support safe, economic and proliferation resistant use of nuclear technology, in a sustainable manner, to meet the global energy needs in the next 50 years and beyond. During Phase I, work is subdivided into two subphases. Phase IA focuses on determining user requirements in the areas of economics, environment, safety, proliferation resistance, and crosscutting issues and developing methodologies and guidelines for the comparison of different reactor and fuel cycle concepts and approaches. The results of Phase IA through June 2003, will be included in a report which is planned to be available at the end of June 2003. Phase IA is now finalised and as of July 2003 Phase IB will start. During this phase interested Member States will perform case studies to assess selected innovative technologies against user requirements using the INPRO methodology and to provide feedback on both – the user requirements and the methodology. In accordance with the INPRO Terms of Reference, after successful completion of Phase I the 2nd Phase of INPRO may be initiated to examine the feasibility of commencing an international project on innovative technology development. The paper contains a description of the background and structure of the INPRO Project.

1. INTRODUCTION

1.1. Establishment of INPRO

The IAEA General Conference (2000) has invited “all interested Member States to combine their efforts under the aegis of the Agency in considering the issues of the nuclear fuel cycle, in particular by examining innovative and proliferation-resistant nuclear technology” (GC(44)/RES/21) and invited Member States to consider to contribute to a task force on innovative nuclear reactors and fuel cycles (GC(44)/RES/22). In response to this invitation, the IAEA initiated the “International Project on Innovative Nuclear Reactors and Fuel Cycles”, INPRO.

At a meeting of senior officials from 25 Member States and international organizations in November 2000, the objectives and conditions of this project were discussed and the Terms of

Reference for INPRO were finalized. INPRO is now being implemented by an International Co-ordinating Group, ICG, for which operating guidelines were adopted. Mr. V. Mourogov, Deputy Director General of the IAEA was appointed as the Project Manager for INPRO. Participants defined the rationale for the project as follows:

"The long-term outlook for nuclear energy should be considered in the broader perspective of future energy needs and environmental impacts. In order for nuclear energy to play a meaningful role in the global energy supply in the foreseeable future, innovative approaches will be required to address concerns about economic competitiveness, safety, waste and potential proliferation risks.

At the national level, work on evolutionary and innovative approaches to nuclear reactor design and fuel cycle concepts is proceeding in several IAEA Member States. At the international level, IEA, OECD/NEA and the IAEA have cooperated in a joint study to review ongoing R&D efforts on innovative reactor designs and to identify options for collaboration. The US Department of Energy is promoting the Generation IV International Forum (GIF) initiative, in which both the IAEA and OECD/NEA are participating as observers. The President of the Russian Federation, at the Millennium Summit, called upon IAEA Member States to join their efforts in creating an innovative nuclear power technology to further reduce nuclear proliferation risks and resolve the problem of radioactive waste."

1.2. Objectives

The objectives of INPRO, as defined in the Terms of Reference, are to:

— Help to ensure that nuclear energy is available to contribute in fulfilling, in a sustainable manner,energy needs in the 21st century;
— Bring together all interested Member States, both technology holders and technology users, to consider jointly the international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles that use sound and economically competitive technology, and are based – to the extent possible – on systems with inherent safety features and minimize the risk of proliferation and the impact on the environment;
— To create a process that involves all relevant stakeholders that will have an impact on, draw from, and complement both the activities of existing institutions and ongoing initiatives at national and international levels.

1.3. Organizational structure

INPRO is an Agency-wide project, with contributions from all relevant IAEA Departments. The framework for implementation of the Project (Table I) consists of the following:

— a Steering Committee (SC), comprising as members, senior officials from Member States that participate through provision of extra-budgetary resources and, as observers, representatives from interested Member States and international organizations. IAEA project management is also represented. The Steering Committee meets as appropriate to provide overall guidance, advise on planning and methods of work and review the results achieved (approximately two times per year);
— an International Co-ordinating Group (ICG), comprising cost free experts from participating Member States, which co-ordinates and implements the project on the basis of experts' work in Member States;

— Technical Expert Groups, comprising experts from Member States, which are convened as appropriate by the ICG to consider specific subjects;

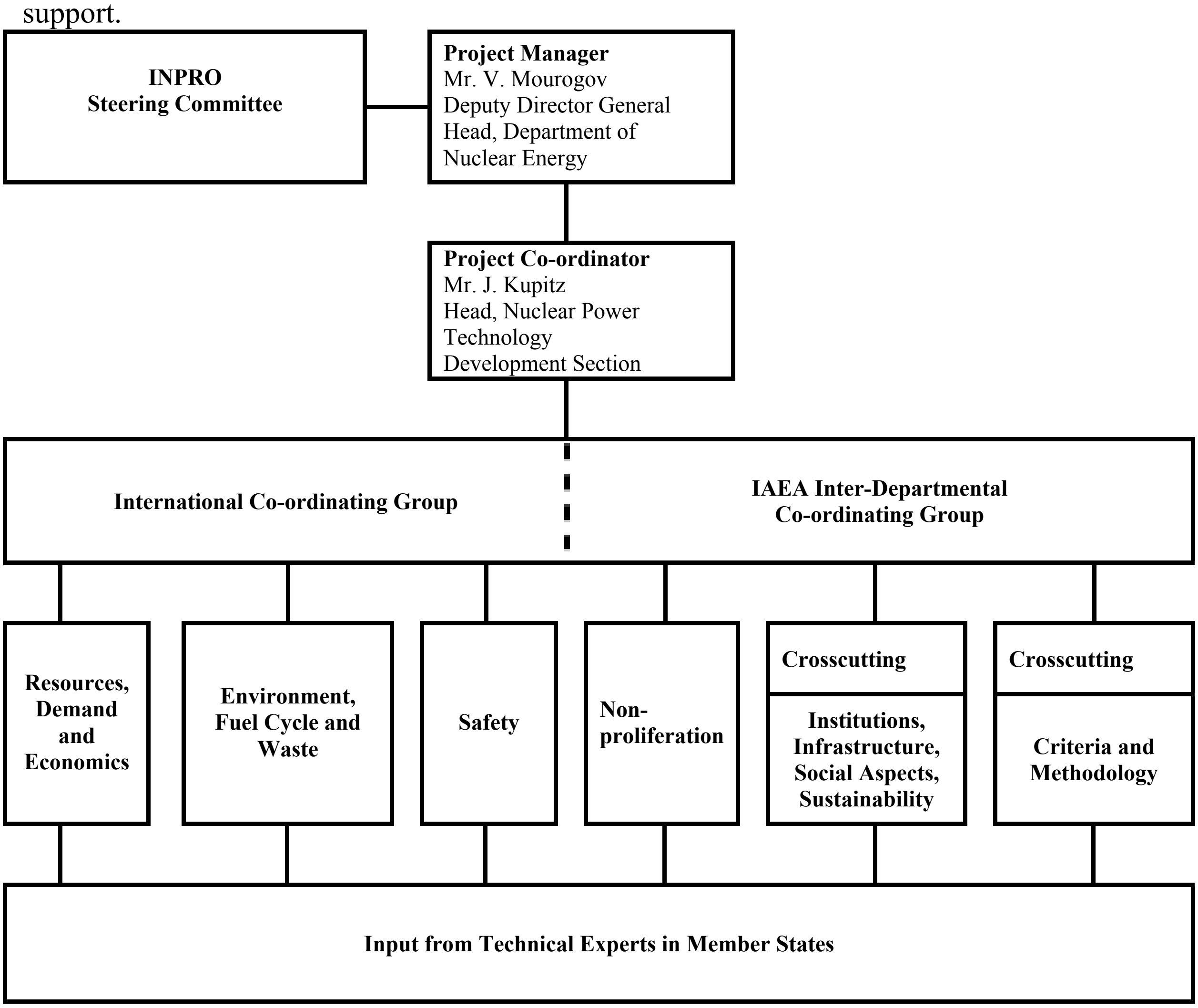
1.4. INPRO membership

As of June 2003, the following countries or entities have become members of INPRO: Argentina, Brazil, Bulgaria, Canada, China, Germany, India, Republic of Korea, Pakistan, Russian Federation, Spain, Switzerland, The Netherlands, Turkey and the European Commission. In total, their respective governments or international organizations have nominated more than 20 cost-free experts.

To become a Member of the INPRO Project a contribution has to be made in the form of either:

— Extrabudgetary funds
— Cost-free Experts
— Work package

Table I. Support from the IAEA, including project management, administrative and technical support.



A work package is a contract between the Agency and an institution in a country, which addresses a specified need within the INPRO project and which represents an effort of at least three man months. The contract is cost-free to the Agency.

1.5. INPRO phases

Phase I of INPRO was initiated in May 2001. During Phase I, work is subdivided in two sub phases:

Phase IA (finalized in June 2003): Selection of basic principles, user requirements and criteria and development of a methodology and guidelines for the comparison of different concepts and approaches.

Phase IB (to be started in July 2003: Performance of Case Studies to validate the INPRO methodology. Three Case Studies have been announced by the following Member States:

- Argentina (Carem)
- India (AHWR)
- Russian Federation (BN 800)

These Case Studies will last for about one year.

Then an examination of innovative nuclear energy technologies made available by Member States against user requirements will be performed. This examination will be co-ordinated by the Agency and performed with participation of Member States based on the user requirements and methodologies established in Phase IA.

In the first phase, six subject groups were established:

- Resources, Demand and User requirements for Economics;
- User requirements for the Environment, Fuel cycle and Waste
- User requirements for Safety;
- User requirements for Non-proliferation;
- User requirements for crosscutting issues;
- Criteria and Methodology.

Upon successful completion of Phase I, taking into account advice from the Steering Committee, and with the approval of participating Member States, Phase II of INPRO may be initiated. Drawing on the results from Phase I, it will be directed to:

- Examining in the context of available technologies the feasibility of commencing an international project;
- Identifying technologies, which might be appropriate for implementation by Member States in such an international project.

1.6. *Co-ordination with other initiatives*

INPRO seeks to interact with other national and international stakeholders and initiatives to ensure effective co-ordination and co-operation in a complementary manner, e.g. with the Generation IV International Forum (GIF) and the OECD. INPRO has already received input from other international organizations; the Three-Agency Study, a study jointly conducted by IEA, OECD/NEA and IAEA on "Innovative Nuclear Reactor Developments – Opportunities for International Co-operation" was provided to INPRO as the joint input of all three participating Agencies. In GIF, the IAEA is represented, as an observer, at the Policy and Experts Group, and IAEA experts participate in the technical meetings of GIF.

The strength of INPRO can be seen in the following main areas:

— *Motivation:* INPRO aims at integrating views from all stakeholders, notably from both nuclear technology developers and nuclear technology end users. User requirements developed with the participation of end users are an essential element in the first phase of INPRO.
— *Time horizon:* The time horizon for INPRO will be very long, and will cover the next five decades. Energy scenarios for the period envisaged will be determined by an expected transformation of the energy sector in the light of limited fossil fuel supplies and potential climate change; new applications such as hydrogen as an energy carrier and seawater desalination for the production of potable water will have to be considered.
— *Scope:* INPRO looks at the whole range of innovative nuclear technologies for both reactors and fuel cycles including the environment, spent fuel and waste, but also institutional aspects and infrastructure. INPRO is aimed at examining the prospects of nuclear technology against this very broad background.
— *IAEA Mandate:* INPRO was initiated through a resolution of the IAEA General Conference and received its mandate from IAEA Member States. In turn, INPRO is established as an open process, and access to results is given to all IAEA Member States.
— *Non-proliferation*: The unique mandate of the IAEA in the area of safeguards helps to ensure that the issue of non-proliferation will be considered at every stage of INPRO.

1.7. *INPRO on the political level*

Following an invitation expressed in two resolutions by the IAEA General Conference in 2000 [GC(44)/RES/21 and GC(44)/RES/22], INPRO was officially launched by the first meeting of the Steering Committee in May 2001.

At the General Conference in 2001, first progress was reported, and the General Conference adopted a resolution on "Agency Activities in the Development of Innovative Nuclear Technology" [GC(45)/RES/12, Tab F], giving INPRO a broad basis of support. The resolution recognized the "unique role that the Agency can play in international collaboration in the nuclear field". It invited both "interested Member States to contribute to innovative nuclear technology activities" at the Agency as well as the Agency itself "to continue it's efforts in these areas".

Additional endorsement came in UN General Assembly resolutions in December 2001 (UN GA 2001, A/RES/56/94) November 2002 (UN GA 2002, A/RES/57/9), that again emphasized "the unique role that the Agency can play in developing user requirements and in addressing safeguards, safety and environmental questions for innovative reactors and their fuel cycles" and stressed "the need for international collaboration in the development of innovative nuclear technology".

INPRO's global character, encompassing both designers and end users and their user requirements, its long time horizon, its consideration of the changing energy sector and its broad based input through IAEA membership make it a valuable forum for the assessment of prospects for nuclear in the 21st century.

1.8. INPRO on the technical level

1.8.1. The innovative nuclear reactors and fuel cycles report

At the first meeting of the Steering Committee in May 2001, the Innovative Nuclear Reactors and Fuel Cycles Report was defined as a key initial product of Phase IA of the project. The Report has the following chapters:

2. CONCLUSIONS

Following an invitation expressed in two resolutions by the IAEA General Conference in 2000, INPRO was officially launched in May 2001.

At the General Conference in 2001, first progress was reported, and the General Conference adopted a resolution on "Agency Activities in the Development of Innovative Nuclear Technology", giving INPRO a broad basis of support. Additional endorsement came in UN General Assembly resolutions in December 2001 and November 2002.

In its Phase IA, INPRO has focused on the establishment of basic principles, user requirements and criteria in various subject areas, and the development of an assessment methodology.

Phase IB will start with various case studies to validate the INPRO methodology.

INPRO is open to all interested IAEA Member States and international organizations; as of June 2003, 15 IAEA Member States or international organizations have become a member.

3. REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENC, TECDOC 1362, International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), IAEA, Vienna (June 2003)

[2] OECD INTERNATIONAL ENERGY AGENCY, OECD NUCLEAR ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Innovative Nuclear Reactor Development Opportunities for International Cooperation, OECD/IEA, Paris (2002)

RESULTS ACHIEVED WITHIN THE INTERNATIONAL PROJECT ON INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES (INPRO)

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Abstract. INPRO has now finalised its Phase 1A. At its last meeting in May 2003 the Steering Committee approved the Final Report[1], which is planned to be available by the end of June 2003. This paper contains a summary of the results achieved within the INPRO project. It reviews the basic principles and user requirements for the areas economics, environment, safety, waste management, proliferation resistance and crosscutting issues and provides a description of the methodology.

1. INTRODUCTION

The main objectives of INPRO are to:

— Help to ensure that nuclear energy is available to contribute in fulfilling energy needs in the 21st century in a sustainable manner; and to
— Bring together both technology holders and technology users to consider jointly the international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles.

In order to set out the boundary conditions for the desired innovations of nuclear energy systems, in Phase 1A, INPRO established several task groups to define:

— Prospects and Potentials of nuclear power within the next 50 years;
— User Requirements for innovative nuclear energy systems (INS) in the area of Economics, Sustainability and Environment, Safety, Waste Management, Proliferation Resistance, and Cross Cutting Issues; and
— Methodology for Assessment of INS.

The results achieved by INPRO as of the end of May 2003 are summarized below [1].

[1] Approved for publication as IAEA TECDOC-1362

2. PROSPECTS AND POTENTIALS

In the area of *Prospects and Potentials* of nuclear power, three topics are briefly discussed: past developments and the current role of nuclear energy, issues surrounding the use of nuclear power, and the potential role of nuclear energy systems in meeting the demand for energy in the 21st century. Early developments in civilian nuclear power were characterized by the need to keep pace with the high energy growth rates of the post-war period, which gave rise to ambitious plans for thousands of GW(e) of nuclear capacity to be installed by the end of the 20th century. But the deployment of nuclear power slowed, primarily because of a decline in the growth of energy demand in the developed countries. Other factors also contributed, such as serious accidents at Three Mile Island and Chernobyl and concerns about the long-term management of spent fuel and high level waste, and about nuclear proliferation.

While expansion of the number of plants has slowed, one very significant recent development has been the steady improvement in availability factors, equivalent to the construction of about 33 new nuclear power plants. The result is that nuclear power has retained its 16% share of global electricity production. Currently, new additions to nuclear capacity are centred in Asia, but signs of revitalization in western Europe and in North America are visible.

The results of a Special Report on Emission Scenarios (SRES), commissioned by the Intergovernmental Panel on Climate Change (IPCC) in 1996, and published in 2000, have been used to examine the expectations and potential for nuclear energy in the 21st century. The SRES presents 40 reference scenarios, grouped according to four storyline families, extending to 2100. Global primary energy grows between a factor of 1.7 and 3.7 from 2000 to 2050, with a median increase by a factor of 2.5 (Figure 1). Electricity demand grows almost 8-fold in the high economic growth scenarios and more than doubles in the more conservational scenarios at the low end of the range. The median increase is by a factor of 4.7. Moreover, nuclear energy plays a significant role in nearly all the 40 SRES scenarios, including the four analysed in INPRO.

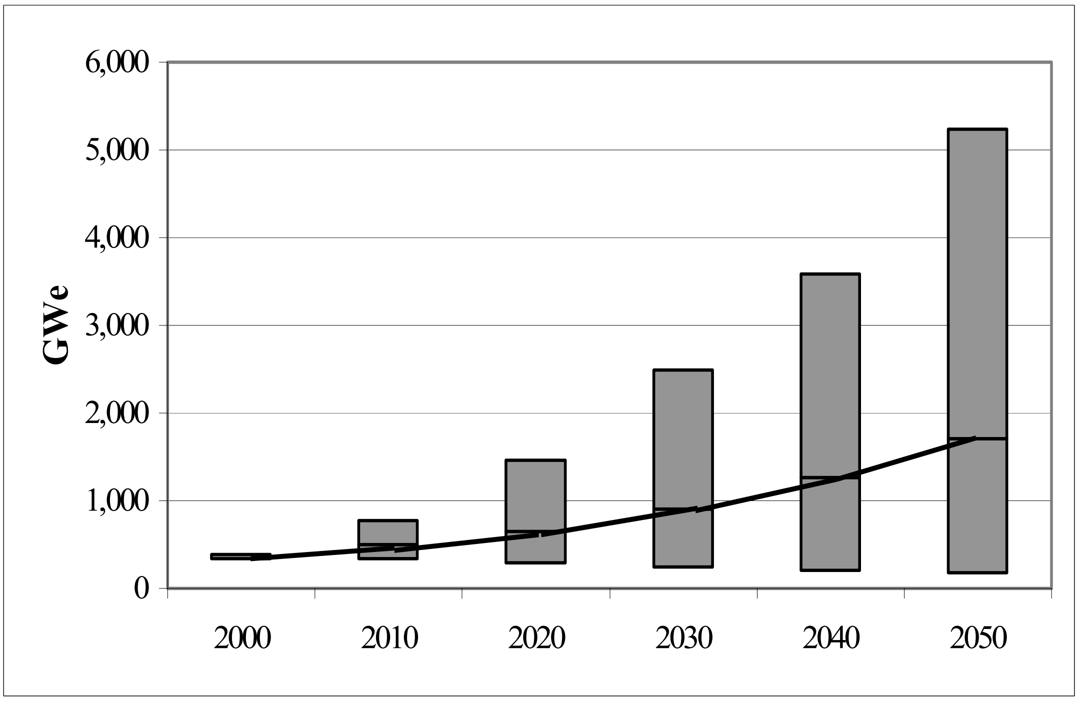


FIG. 1. Range of nuclear power in SRES scenarios, 2000-2050. Solid line represents median.

This contrasts with near-term projections by the IAEA, OECD/IEA and US DOE Energy Information Administration that show a declining nuclear share in global electricity production in coming decades, and little or no nuclear movement into energy applications beyond electricity. The difference between these more pessimistic near-term projections and a truly substantial future contribution of nuclear energy – one that takes nuclear's percentage of the world's primary energy supply well beyond today's single digits to 20%, 50% or more – is innovation. Innovation represents the driving force for continuous development of nuclear technologies leading to INS that will be superior to existing plants. These systems comprise not only electricity generating plants, but they also include, e.g., plants (of various size and capacity) for high-temperature heat production, district heating and sea water desalination, to be deployed in developed regions as well as in developing countries and countries in transition.

INS therefore can play an important role in meeting this rapidly expanding world energy demand, consistent with the principle of sustainable development, i.e. meeting the needs of current generations without compromising the ability of future generations to meet their needs. To achieve this objective the issues on which debate concerning the future role of nuclear energy is most often focused need to be addressed. These issues are: economic competitiveness, safety, waste management, proliferation resistance and physical protection, and last, but not least, sustainability and environment.

INPRO has examined the needs to be met by innovative nuclear energy systems in each of these areas and has defined a set of *Basic Principles, User Requirements*, and *Criteria* (consisting of an *Indicator* and an *Acceptance Limit*) for each area. *Users* encompass a broad range of groups including investors, designers, plant operators, regulatory bodies, local organizations and authorities, national governments, NGOs and the media, and last not least the end users of energy (e.g., the public, industry, etc).

3. ECONOMICS

In the area of Economics four selected scenarios from the SRES study have been analyzed (Figure 2). They cover a variety of possible future developments that are characterized by differing levels of globalisation and regionalization and by differing views of economic growth versus environmental constraints.

Provided INS are economically competitive they can play a major role in meeting future energy needs. Economic competitiveness depends on the learning rates (cost reductions as a function of experience) achieved by nuclear energy relative to those of competing technologies. Specific capital costs and electricity production costs have been derived, which are indicative of costs that would enable nuclear energy to compete successfully against alternative energy sources for the four marker scenarios chosen (Figure 3). These costs should be used with caution since they depend on the learning rates for competing technologies implicit in the SRES scenarios, the discount rates used, and the fact that risks are not taken into account.

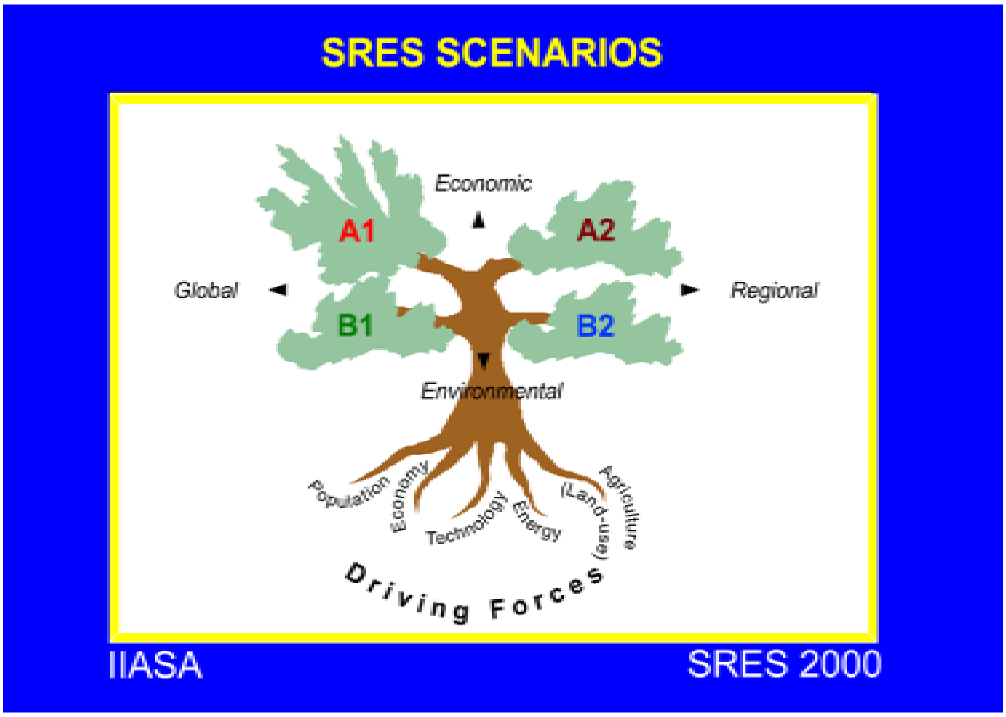
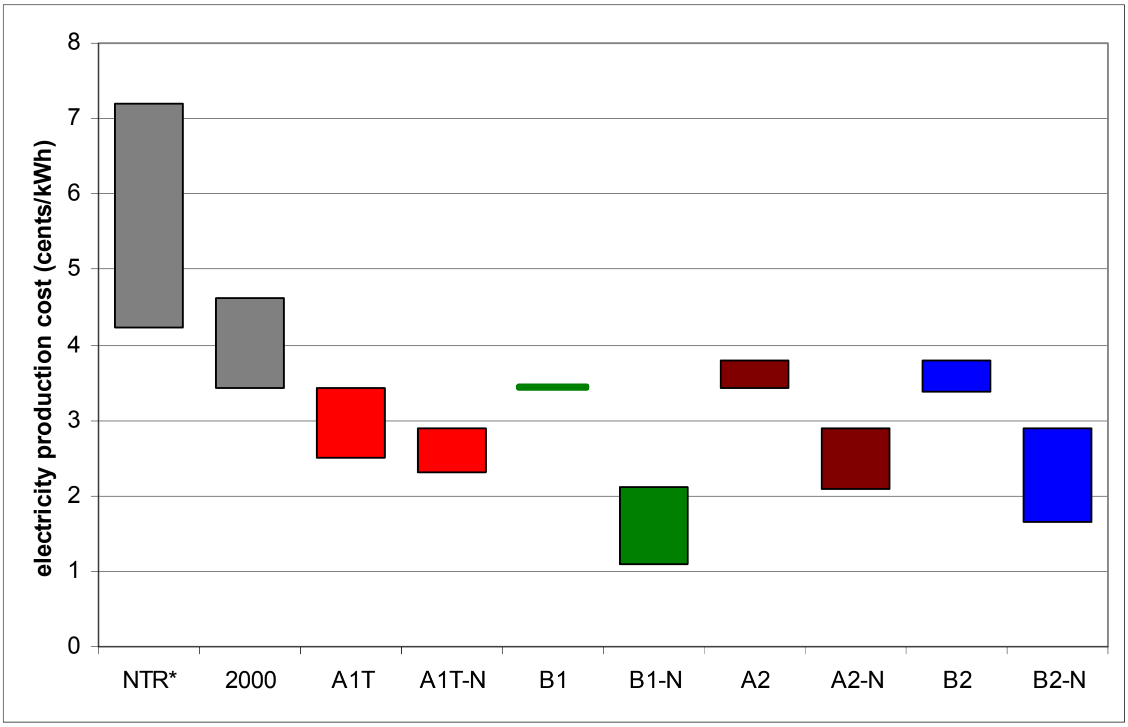


FIG. 2. Schematic illustration of the four SRES storyline families.



NTR* = Nuclear Technology Review (IAEA)
2000 = SRES Input in the year 2000

FIG. 3. Ranges for electricity production costs (exclusive of fuel costs) in 2050 for nuclear power plants in eight scenarios.

The important message is that for nuclear technology to gain and grow market share it must benefit sufficiently from learning to keep it competitive with competing energy technologies. For such learning to take place experience must be gained, the energy from INS must remain cost competitive with energy from alternative sources, and INS must represent an attractive investment to compete successfully in the capital market place.

To be cost competitive all component costs, e.g., capital costs, operating and maintenance costs, fuel costs, must be considered and managed to keep the total unit energy cost competitive. Limits on fuel costs in turn imply limits on the capital and operating cost of fuel cycle facilities, including mines, fuel processing and enrichment, fuel reprocessing and the decommissioning and long-term management of the wastes from these facilities. Cost competitiveness of energy from INS will contribute to investor confidence, i.e. to the attractiveness of investing in INS, as will a competitive rate of return (see Table I for an example of the corresponding Basic Principle, User Requirements and Criteria). As well, meeting the *Principles* and *Requirements* established by INPRO in the areas of safety, waste management, sustainability, and proliferation resistance will also contribute to investor confidence.

4. SUSTAINABILITY AND ENVIRONMENT

Internationally there exists strong interest and support for the concept of sustainability, as documented in the report of the Bruntland Commission, the Rio declarations, etc. There is a prima facie case that nuclear power supports sustainable development by providing much needed energy with relatively low burden on the atmosphere, water, and land use. Further deployment of nuclear power would help to alleviate the environmental burden caused by other forms of energy production, particularly the burning of fossil fuels. INPRO has set out two Basic Principles related to *Sustainability*, one dealing with the acceptability of environmental effects caused by nuclear energy and the second dealing with the capability of INS to deliver energy in a sustainable manner in the future. Protection of the environment from harmful effects is seen to be fundamental to sustainability. Adherence to the principle that the present generation should not compromise the ability of future generations to fulfil their needs, requires that the future be left with a healthy environment. Notwithstanding the major environmental advantages of nuclear technology in meeting global energy needs, the potential adverse effects that the various components of the nuclear fuel cycle may have on the environment must be prevented or mitigated effectively to make nuclear energy sustainable in the long term. Environmental effects include: physical, chemical or biological changes in the environment; health effects on people, plants and animals; effects on quality of life of people, plants and animals; effects on the economy; use/depletion of resources; and cumulative effects resulting from the influence of the system in conjunction with other influences on the environment. Both radiological and non-radiological effects as well as trade-offs and synergies among the effects from different system components and different environmental stressors need to be considered.

Table I. Example for a User Requirement and Criteria related to Economic Principle 2 (In total, INPRO has defined 2 Principles and 5 User Requirements for Economics).

Basic Principle 2: Innovative Nuclear Energy Systems must represent an attractive investment compared with other major capital investments

User Requirement	Criteria	
	Indicator	Acceptance Limit
The total investment required to design, construct, and commission innovative nuclear energy systems, including interest during construction, must be such that the necessary investment funds can be raised.	Total investment.	Investment in INS enable a return comparable with or better than that required to deploy a competing energy technology of comparable size.
	Project construction and commissioning times.	Times comparable to alternative projects. Schedules met.

To be sustainable the system must not run out of important resources part way through its intended lifetime. These resources include fissile/fertile materials, water (when supplies are limited or quality is under stress) and other critical materials. The system should also use them at least as efficiently as acceptable alternatives, both nuclear and non-nuclear (see Table II for an example of the corresponding Basic Principles, User requirements and Criteria).

Table II. Example for a User Requirement and Criteria related to Basic Principle 2 of Sustainability (In total, INPRO has defined 2 Principles and 4 User Requirements for Sustainability).

Basic Principle 2: Fitness for Purpose
The innovative nuclear energy system must be capable of contributing to energy needs in the future while making efficient use of non-renewable resources.

User Requirements	Criteria	
	Indicator	Acceptance Limit
The system should be able to meet a significant fraction of the world's energy needs during the 21st century without running out of fissile/fertile material and other non-renewable materials, with account taken of reasonably expected uses of these materials external to the energy system.	F_{ci}: Fuel i consumed in 100 yrs (Mg).	$F_{ci} \leq (F_{pr-i}+ F_{ri})$; F_{pr-i} : Fuel i proven reserves (Mg), and $F_{ri;}$ Fuel i reprocessed in 100 yrs (Mg).
	M_{ci}: Critical material i consumed in 100 yrs (Mg).	$M_{ci} \leq M_{pri;}$ M_{pri} : Proven reserves of critical material i (Mg)
	$B_{up} = E / U$; B_{up} : burnup. E: provided energy (MWd). U: consumed fissile material (Mg).	$B_{up} > B_{up}\,Ref$ $B_{up}\,Ref$: reference burnup.

All relevant factors (sources, stressors, pathways, receptors and endpoints) must be accounted for in the analysis of the environmental effects of a proposed energy system, and the environmental performance of a proposed technology needs to be evaluated as an integrated whole by considering the likely environmental effects of the entire collection of processes, activities and facilities in the energy system at all stages of its life cycle.

5. SAFETY OF NUCLEAR INSTALLATIONS

In the area of *Safety of Nuclear Installations*, INPRO recognizes that extensive work has been done prior to INPRO to establish safety requirements included in documents such as the Advanced Light Water Reactor Utility Requirements prepared by EPRI, the European Utility Requirements prepared by European Utilities, IAEA Safety Standards Series, e.g., Safety Guides, and INSAG documents. The safety Principles and Requirements developed within INPRO are based on extrapolation of current trends and seek to encompass the potential interests of developing countries and countries in transition. For nuclear reactors, the fundamental safety functions are to control reactivity, remove heat from the core, and confine radioactive materials and shield radiation. For fuel cycle installations, they are to control sub-criticality and chemistry, remove decay heat from radionuclides, and confine radioactivity and shield radiation. To ensure that INS will fulfil these fundamental safety functions, INPRO has set out five Basic Principles but it is also expected that prior work will also be used to the extent applicable.

INPRO expects that INS will incorporate enhanced defence-in-depth as part of their basic approach to safety but with more independence of the different levels of protection in the defence-in-depth strategy, and with an increased emphasis on inherent safety characteristics and passive safety features. The end point should be the prevention, reduction and containment of radioactive releases to make the risk of INS comparable to that of industrial facilities used for similar purposes so that for INS there will be no need for relocation or evacuation measures outside the plant site, apart from those generic emergency measures developed for any industrial facility (see Table III for an example of the Basic Principles, User Requirements and Criteria for Safety).

Table III. Example for a User Requirements and Criterion related to Basic Principle 1 for Safety (In total, INPRO has defined 5 Basic Principles and 27 User Requirements for Safety).

Basic Principle 1: *Innovative nuclear reactors and fuel cycle installations shall incorporate enhanced defence-in-depth as a part of their fundamental safety approach and the levels of protection in defence-in-depth shall be more independent from each other than in current installations.*

User Requirement	Criteria	
	Indicators	Acceptance Limit
The innovative nuclear reactors and fuel cycle installations shall not need relocation or evacuation measures outside the plant site, apart from those generic emergency measures developed for any industrial facility.	Probability of large release of radioactive materials to the environment.	$<10^{-6}$ per plant*year, or excluded by design.

RD&D must be carried out before deploying INS, using e.g., large scale engineering test facilities including, possibly, pilot plants, to bring the knowledge of plant characteristics and the capability of codes used for safety analyses to the same level as for existing plants. The development of INS should be based on a holistic life cycle analysis that takes into account the risks and impacts of the integrated fuel cycle. Safety analyses will involve a combination of deterministic and probabilistic assessments, including best estimate plus uncertainty analysis.

6. WASTE MANAGEMENT

Because *Waste Management* involves longer time scales and, in many cases, different source terms and pathways, compared with nuclear installations, this topic is dealt separately from the safety of nuclear installations. The already existing nine principles defined by the IAEA for the management of radioactive waste have been adopted by INPRO without modification. Thus, waste management is to be carried out in such a way that human health and the environment are protected now and in the future, effects beyond national borders shall be taken into account, undue burdens passed to future generations shall be avoided, waste shall be minimized, appropriate legal frame works shall be established and interdependencies among steps shall be taken into account. These principles in turn lead to INPRO requirements to specify a permanently safe end state(s) for all wastes and to move wastes to this end state as early as practical, to ensure that intermediate steps do not inhibit or complicate the achievement of the end state, that the design of waste management practices and facilities be optimised as part of the optimisation of the overall energy system and life cycle, and for assets to cover the costs of managing all wastes in the life cycle to be accumulated to cover the accumulated liability at any stage of the life cycle (See Table IV for an example of the User Requirements and Criteria).

Table IV. Example for a User Requirement (adverse effects on the environment) and Criteria for Safety of Waste Management (In total, INPRO has defined 9 Basic Principles and 6 User Requirements).

Basic Principle 2: Radioactive waste shall be managed in such a way as to provide an acceptable level of protection of the environment.

User Requirement	Criteria	
	Indicators	Acceptance Limit
Adverse Effects on the Environment: Waste management strategies should be such that the adverse environmental effects from all parts of the energy system and the complete life cycle of facilities are optimized. The cumulative effects over time and space, without regard to national boundaries, should be considered.	Estimated concentrations of radionuclides and chemical toxins in the environment.	Meet standards of specific Member State.
	Exposures of sensitive species to these expected concentrations.	Would not be expected, on a scientific basis, to cause adverse effects at the population level.
	Other environmental indicators.	Meet requirements as specified in Task 4.2 of INPRO.

It is also expected that prior work carried out by the IAEA in waste management will be used to the extent possible. RD&D is recommended to be carried out in a number of areas including partitioning and transmutation and long term human factors analysis to facilitate assessments of long term risks for waste management systems that require long term institutional controls.

7. PROLIFERATION RESISTANCE

In designing future nuclear energy systems, it is important to consider the potential for such systems being misused for the purpose of producing nuclear weapons. Such considerations are among the key considerations behind the international non-proliferation regime a fundamental component of which is the IAEA safeguards system. INPRO set out to provide guidance on incorporating *Proliferation Resistance* into INS. The INPRO results in this area are largely based on the international consensus reached in October 2002 at a meeting held in Como, Italy. Generally two types of proliferation resistance measures or features are distinguished: intrinsic and extrinsic. Intrinsic features result from the technical design of INS including those that facilitate the implementation of extrinsic measures. Extrinsic measures are based on States' decisions and undertakings related to nuclear energy systems.

Intrinsic features consist of technical features that: a) reduce the attractiveness for nuclear weapons programmes of nuclear material during production, use, transport, storage and disposal, including material characteristics such as isotopic content, chemical form, bulk and mass, and radiation properties; b) prevent or inhibit the diversion of nuclear material, including the confining of nuclear material to locations with limited points of access, and materials that are difficult to move without being detected because of size, weight, or radiation; c) prevent or inhibit the undeclared production of direct-use material, including reactors designed to prevent undeclared target materials from being irradiated in or near the core of a reactor; reactor cores with small reactivity margins that would prevent operation of the reactor with undeclared targets; and fuel cycle facilities and processes that are difficult to modify; and d) that facilitate nuclear material accounting and verification, including continuity of knowledge. Five categories of extrinsic features are defined, as follows: commitments, obligations and policies of states, such as the Treaty on the Non-Proliferation of Nuclear Weapons and the IAEA safeguards agreements; agreements between nuclear material exporting and importing states; commercial, legal or institutional arrangements that control access to nuclear material and technology; verification measures by the IAEA or by regional, bilateral and national measures; and legal and institutional measures to address violations of measures defined above.

INPRO has produced Basic Principles that require: the minimization of the possibilities of misusing nuclear material in INS; a balanced and optimised combination of intrinsic features and extrinsic measures; the development and implementation of intrinsic features; and a clear, documented and transparent method of assessing proliferation resistance. To comply with these Basic Principles requires the application of the concept of defence-in-depth by, e.g., incorporating redundant and complementary measures; an early consideration of proliferation resistance in the development and design of INS; and the utilization of intrinsic features to increase the efficiency of extrinsic measures (see Table V for an example of a User Requirement and Criterion). RD&D is needed in a number of areas, in particular, in developing a process to assess the proliferation resistance of a defined INS.

Table V. Example for a User Requirement and Criterion for Proliferation Resistance (In total, INPRO has defined 5 Basic Principles and 5 User Requirements for Proliferation Resistance).

User Requirement	Criterion	
	Indicator	Acceptance Limit
The combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimized to provide cost-effective proliferation resistance.	Cost for incorporating intrinsic features and applying extrinsic measures required to provide adequate proliferation resistance.	Minimal Cost.

8. CROSS CUTTING ISSUES

Issues other than technical requirements are important to potential users of INS. Many of the factors that will either facilitate or obstruct the on-going deployment of nuclear power over the next fifty years are *Cross Cutting Issues* that relate to nuclear power infrastructure, international cooperation, and human resources. Nuclear power infrastructure comprises all features/ substructures that are necessary in a given country for the successful deployment of nuclear power plants including legal, institutional, industrial, economic and social features/substructures. The SRES scenarios indicate that the growth of nuclear power will be facilitated by globalization and internationalization of the world economy, and that the growth of demand in developing countries will be a major consideration. Globalization and the importance of developing countries in future world energy markets point to the need to adapt infrastructures, both nationally and regionally, and to do so in a way that will facilitate the deployment of nuclear power systems in developing countries.

In a globalizing world with a growing need for sustainable energy, harmonization of regulations and licensing procedures could facilitate the application of nuclear technology. Such harmonization among different markets is in the interest of suppliers and developers of technology as well as users and investors. The development of innovative reactors to comply with the Basic Principles, User Requirements and Criteria set out in this report should facilitate such harmonization and could make it possible to change the way the production of nuclear energy is regulated. When, for example, the risk from INS are ‘comparable to that of industrial facilities used for similar purposes,’ and ‘there is no need for relocation or evacuation measures outside the plant site, apart from those generic emergency measures developed for any industrial facility,’ the requirements for licensing could possibly be simplified. In developing countries, and amongst them countries that do not have a highly developed nuclear knowledge base and infrastructure, the development of regional or international licensing and regulatory mechanisms and organizations could play an important role. Additional factors that would be expected to favour the deployment of INS, particularly in developing countries include: optimisation of the overall nuclear energy system by considering component facilities located in different countries as part of an international multi-component system; recognizing the needs of developing countries that have a limited infrastructure and a real but limited need for nuclear energy; vendor countries offering a full-scope service, up to and including the provisions of management and operations.

The life cycle of nuclear power systems, including design, construction, operation, decommissioning, and the waste management, extends well over fifty years in most cases and can easily extend well beyond one hundred years. Thus, a firm long-term commitment of the government and other stakeholders is seen as a requirement for the successful implementation and operation of a nuclear power investment and a condition for public acceptance. Clear communications on energy demands and supply options are important to developing an understanding of the necessity for and the benefits to be obtained from such long-term commitments. A clear enunciation of the potential role of nuclear energy in addressing climate change concerns in a sustainable and economic manner, together with the performance of existing plants can play an important role in such communications.

The development and use of nuclear power technology requires adequate human resources and knowledge. Globalization brings with it the opportunity to draw on a much broader pool of resources rather than striving to maintain a complete domestic capability across the many disciplines of science and engineering that constitute the range of technologies on which nuclear energy systems depend. International cooperation in science and development can assist with optimizing the deployment of scarce manpower and, just as important, the construction and operation of large-scale research and engineering test facilities.

9. METHODOLOGY FOR ASSESSMENT

INPRO has also developed a methodology for evaluating INS, the INPRO *Methodology*. It comprises the INPRO Basic Principles, User Requirements, and Criteria, and a set of tables and guidance on their use, that can be used to evaluate a given innovative energy system, or a component of such a system on a national, regional and/or global basis. The INPRO Methodology is oriented more to identifying a range of technology alternatives that will fulfil Basic Principles and User Requirements set out for INS, rather than to selecting a single best solution. It is recognized that the methodology will need to be applied iteratively, that the INPRO User Requirements and Criteria may be supplemented by additional Requirements and Criteria, e.g., taken from existing Standards and Guides, and that additional work is likely required to elaborate requirements and standards. To assess a given nuclear energy system (or a component thereof) the nuclear energy system and its components are specified together with approaches for meeting all relevant Criteria, User Requirements and Basic Principles. Judgments are then established of the potential of the approaches and their constituent components to meet the Criteria, User Requirements and Basic Principles for the nuclear energy system, and a Judgment of the entire system is arrived at from the Judgments for compliance with all of the Basic Principles, User Requirements, and Criteria. Member States (MS) must identify all of the fuel cycle components that will be required for the MS to use the component of prime interest to it, e.g., a given design of reactor, and present information on all components so that a holistic view is developed and presented. The rationale for arriving at a given Judgment, i.e. the basis of the Judgment, needs to be developed and explained. The rationale may be based, e.g., on preliminary or detailed safety and environmental analyses, experience with large-scale test facilities or experimental test rigs, extrapolation of experience from similar facilities, the use of expert opinion, and combinations of these. Additional effort will be needed to develop the methodology further for widespread use and to ensure consistency and credibility of the results. Prior to committing to such an effort an assessment of the efficacy of the methodology should be obtained by using it in a number of case studies. It is foreseen that case studies will be performed by individual interested MS supported by task groups with broader participation of experts from MS of INPRO. To test the methodology, case studies will be carried out for different types of nuclear energy systems, including a global system with components at the preliminary stage of development, a future system that is already reasonably well developed, and systems being considered for application in different regions.

10. CONCLUSIONS AND RECOMMENDATIONS

Phase 1A was an important first step toward INPRO's two objectives of (1) ensuring the availability of nuclear energy to contribute to meeting growing global energy needs in the 21st century and (2) bringing together prospective buyers and sellers of nuclear technology, nuclear "haves" and "have-nots", and developing and developed countries to jointly consider actions needed to accelerate nuclear innovation in directions most likely to be most useful to the energy markets of the future.

Phase 1A has reviewed expected energy needs in the 21st century, and the potential role of nuclear energy, using scenarios from the Special Report on Emission Scenarios by the Intergovernmental Panel on Climate Change. SRES clearly shows that energy demand, and especially electricity demand, will grow substantially regardless of which mix of driving forces ends up dominating future world developments. Moreover, nuclear energy plays a significant role in nearly all the 40 SRES scenarios, including the four analyzed in this paper. This contrasts with near-term projections by the IAEA, OECD-IEA and US DOE Energy Information Administration that show a declining nuclear share in global electricity production in coming decades, and little or no nuclear movement into energy applications beyond electricity. The difference between these more pessimistic near-term projections and a truly substantial future contribution of nuclear energy – one that takes nuclear's percentage of the world's primary energy supply well beyond today's single digits to 20%, 50% or more – is innovation. The pathway to this future is innovative nuclear energy systems.

The 21st century promises the most competitive, globalized markets in human history, the most rapid pace of technological change ever, and the greatest expansion of energy use, particularly in developing countries. For a technology to make a truly substantial contribution to energy supplies, innovation is essential. It will be the defining feature of a successful nuclear industry and a critical feature of international co-operation in support of that industry, cooperation that ranges from joint scientific and technological initiatives, to safety standards and guidelines, and to security and safeguards activities. Innovation is also essential to attract a growing, high-quality pool of talented scientists, engineers and technicians of the calibre and size needed to support a truly substantial nuclear contribution to global energy supplies.

To help co-ordinate and guide the development of innovative nuclear energy systems, INPRO Phase 1A has set out initial Basic Principles, User Requirements and corresponding Criteria in the areas of economics, the environment, safety, waste management, and proliferation resistance. Cross-cutting issues related to infrastructure and international co-operation have also been discussed. A methodology for assessing innovative nuclear energy systems has been created for the use of Member States and independent analysts. It complements and builds upon requirements and criteria set out in existing documents such as the IAEA Safety Standards Series. All these outputs, from basic principles to the INPRO assessment methodology, are expected to be steadily sharpened and adjusted based on feedback from early applications and case studies.

Specific recommendations for the future are that:

- INPRO be continued, and that co-operation and co-ordination between INPRO and other initiatives on innovative nuclear energy systems be strengthened;
- As part of a consecutive phase, called Phase 1B of INPRO, Member States define in further detail the RD&D initiatives set out in the report and set out priorities. The IAEA could provide valuable assistance in facilitating co-operation among Member States and establishing complementary Co-ordinated Research Programs;

— Case Studies be encouraged to enable Member States and independent analysts to assess prospective innovative nuclear energy systems using the INPRO methodology; and
— Feedback and experience from Case Studies and other applications be used to sharpen and adjust the INPRO Basic Principles, User Requirements, Criteria and Methodology to continually improve their usefulness.

11. ACKNOWLEDGEMENTS

The IAEA highly appreciates the guidance and advice received from all experts who participated in INPRO meetings.

12. REFERENCES

[1] IAEA TECDOC 1362 International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), June 2003.

U.S.–RUSSIAN COLLABORATIVE PROGRAM

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Abstract. The paper considers the experience of the last decade, which already demonstrates promising examples of joint Russian-U.S. activities aimed at developing collaboration in innovative reactor and fuel cycle area. The paper discusses the consequences of 2002 “breakthrough” in Russian-U.S. cooperation after the May Summit of the two countries’ Presidents, as well as the challenges faced by this cooperation, and its further perspectives. The paper notes the role of bilateral initiatives on non-governmental level in the field of innovative reactor and fuel cycle technologies.

1. INTRODUCTION

United States and Russia are the countries, whose nuclear programs have long been an example for others in nuclear technology development. “Atoms for Peace” initiative put forward by President Eisenhower in 1953, and the first nuclear power plant created by Academician Kurchatov in 1954, have marked the beginning of the first nuclear era.

Before the historical landmark of 1955, nuclear programs in the United States and Soviet Union were developing almost independently. Nevertheless, according to the witnesses, the specialists gathered at the 1st International Conference for Peaceful Use of the Atomic Energy in Geneva have been surprised to find out, that not only the ideas, but also many technical solutions independently found by scientists and engineers from various countries, were common.

Opening of top-secret works in the field of nuclear energy has undoubtedly made the desirable international collaboration possible in principle. However, the next 20-25 years cannot be called an active period. Nuclear power industries of West and East developed with quite limited contacts, separated by the “Berlin Wall” – through which, nevertheless, successes and difficulties of the neighbour could be observed and good ideas and solutions could be sometimes adopted.

This wasn’t a one-way street. There were fields, where Soviet nuclear specialists had much to show to the world, and both the mutual influence and the trace of Soviet achievements in the U.S. nuclear programs can be easily seen. For example, such areas include centrifugal separation of isotopes, nuclear submarine fleet and space nuclear power.

2. EXPERIENCE OF THE LAST DECADE

The last decade already shows promising examples of joint activities aimed at developing cooperation in the area of innovative reactors and fuel cycles.

They include the "breakthrough" in the space nuclear power collaboration, where – besides the Geneva conferences – the first contacts between specialists took place in 1989 and rapidly developed into the joint Russia-U.S. program of testing the prototype Russian space nuclear installation "Topaz-2" on test facilities (1992-1993). Joint works of Russian and U.S. organizations and companies in the field of thermoemission converters still continue today.

Another innovative project area, where the joint U.S.-Russian initiative has developed into an active international program with participation of French and Japanese companies and objective prerequisites for successful realization, is the development of high-temperature helium-cooled gas-turbine reactor (GT-MHR). The program is based on the technical solutions approbated in the United States in course of HTGR construction and operation, and on the 30 years of Russian experience in designing pilot industrial facilities, developing high-temperature fuel elements and main equipment.

Considerable bilateral activities in the field of civil nuclear reactor safety (framed in the end of the 80ies by the similarly named commission, in which leading roles were played by NRC, from one part, and by the Kurchatov Institute and then a wide circle of Minatom organizations, from another), should be considered as a basic element for the development of innovative collaboration. This collaboration (quite broad and inevitable after Chernobyl) was conducted in various forms, including deep studying of one country's NPP designs by another country's specialists, and even the adoption of advanced technology – as, for instance, reactor vessel restoration born in the Russian power industry.

Bilateral contacts within large international projects have always been – and still are – very fruitful. One of these projects was OECD/NEA's "Rasplav' project (presently MASCA) – the program of extreme severe accident experiments, which, with Russian leadership, involve very important U.S. participation. Another example – international criticality safety benchmark evaluation project, initiated by the United States also under OECD/NEA auspices, in which the Russian contribution is now strengthening. By the way, such projects often grow into bilateral programs – for example, a series of precision safety experiments in IPPE and the Kurchatov Institute ordered by INEEL and with its participation.

On the whole, all the above and many other developments have formed a solid base for a new step in Russian-U.S. nuclear power cooperation, when the political will was added to the historical experience.

The initiative of ensuring energy supply of the sustainable development of mankind put forward by the Russian President at the Millennium Summit in 2000 was based on the obvious end of stagnation in the Russian nuclear power industry and definitely influenced the world nuclear power process. The U.S. national energy policy declared a year ago was based on similar ideas. This new energy policy recognized nuclear energy as a factor of stability in economic development and an environmentally acceptable part of the energy mix option.

It should be noted that the new element of the U.S. nuclear policy, which withdrew the ban at least for R&D related to nuclear fuel reprocessing, has left in the past the U.S.-Russian "dissidence" in the principal innovative nuclear fuel cycle area.

3. “BREAKTHROUGH”-2002

Obvious community of both Presidents’ positions has developed into concrete decisions during their meeting in May’2002.

According to the data of the Russian Ministry of Atomic Energy, the “starting points” for the new phase of cooperation was represented by the following collaboration agreements in the area of peaceful atom:

— nuclear waste storage facility construction at the Russian territory;
— weapon-grade uranium depletion into low-enriched one for the U.S. NPPs;
— utilization of waste from Russian nuclear submarine reactors;
— works on nuclear process and material research in the framework of the International Research Center;
— contract for using the Russian Pu-238 isotope in spacecraft engines.

The meeting was preceded by the joint initiative of the two national laboratories – U.S. Sandia National Laboratories and Russian Research Centre “Kurchatov Institute”, which have prepared and published a joint report entitled “The Global Nuclear Future: from Atoms for Peace to Atoms for Peace and Prosperity”. The paper addressed the energy and environmental challenges of the XXI century, energy situation in the USA and Russia, as well as the lessons learned by the two countries from the “first nuclear era”.

The main conclusion of the report is that the current phase of nuclear power development has confirmed its viability. The key problems were identified, and advanced concepts may become a key for global stability. Principal ways to solve these problems are already known, and after their realization in the current century the beginning of the new phase in nuclear energy use – the large-scale nuclear power development – will become possible.

Then the nuclear energy founder states may once again become main nuclear technology suppliers to the developing countries, which have no energy resources of their own.

“The accumulated experience in the U.S. and Russia is the basis of the feasibility of large-scale global nuclear power development. Now is the time for the two former nuclear leaders to develop a joint initiative that will define the new nuclear era and assure the extended use of nuclear technologies will indeed contribute to global energy security, as well as to the improvement of health, well being, and living and environmental conditions for all people”, - says the report.

The authors saw their address to the leaders of the two countries reflected in the joint statement of the two Presidents on the “new U.S.-Russian energy dialogue”, in which the parties have announced their intention “to collaborate in developing new, more environmentally safe nuclear power technologies”. The first step towards realizing this intention consisted, in accordance with the Joint Declaration of “the new strategic relations”, in setting up joint expert groups, including the group for “preparing recommendations on joint efforts in research and development related to advanced nuclear reactor and fuel cycle technologies”.

This group of governmental experts headed by the U.S Under Secretary of Energy Robert Card and the First Deputy Minister of Atomic Energy of Russia Lev Ryabev worked in period of June-July 2002, and submitted its report at the conclusion.

It should be noted that this work contributed to the experience of supporting the governmental decision undergoing the preparatory process by the activities of non-governmental experts. The well-known report of "Velikhov-Holdren" U.S.-Russian independent scientific commission (1997), after which the United States have supported the Russian initiative to use plutonium in MOX fuel form for burning in nuclear power reactors, can be considered as an efficient start of this process.

In our case, in parallel with the work of DOE-Minatom group, a meeting of non-governmental experts from Russia and USA: "The Future of Nuclear Power: Energy, Ecology, Safety" organized by the Kurchatov Institute with support of the Russian Academy of Sciences and the Nuclear Society of Russia, as well as of MIT, Carnegie and Scowcroft Foundations from the U.S. part, has been conducted.

An important result of this forum's work was represented by practically first-proposed targets of global and regional nuclear power development till the middle of the century: it was proposed "to consider and discuss increasing by the mid-century the nuclear energy generation level 4-5-fold compared to the present-day one" … "as a goal, which is sufficiently important to influence, on the global scale, the electricity production, energy security and reducing the greenhouse effect".

The following scenario of regional distribution of nuclear power capacities by 2050 was proposed: 1000 GWe in the United States, Europe and the developed countries of the Eastern Asia, 100 GWe – in the former-USSR countries, and 400 GWe – in the developing world.

Large-scale international collaboration is an obvious prerequisite for realization of this scenario. The United States and Russia, with account of the amount of nuclear materials produced, should especially "deeply" collaborate in further developing the fuel cycle and implementing the non-proliferation regime.

Proposed division of nuclear power development into periods distinguishes its current phase, when in the next decade any considerable changes both in the amount of nuclear-generated electricity and in the character of the used technologies. Nevertheless, this is a period of development and commissioning of advanced reactor designs (LWR, HTGR, small reactors), which will form the base of the next phase of nuclear power capacities' growth.

Several next decades (till the mid-century) will be the initial period of growth, when many-fold extension of nuclear power scale will be required. This will be followed by the period of deploying principally new technologies, which will inevitably require realization of the closed fuel cycle. In all probability, thermal (light-water, high-temperature gas-cooled, molten salt) and fast (liquid metal, gas-cooled) reactors with closed U-Pu and Th-U fuel cycles will be functioning in the power system during this period.

Here the United States and Russia should unite their efforts in developing new reactor technologies and fuel cycle of the future, which "could be deployed in the developing countries" by introducing fuel and reactor leasing, as well as complete operation "cradle-to-grave" contracts. In the experts' opinion, a limited number of large nuclear fuel cycle centres should be established for supplying nuclear power reactors and fuel to the customer countries and returning spent nuclear fuel for reprocessing, conditioning and waste management.

The ideas of the 2002 expert meeting gave a certain impetus to preparing a bilateral report on Russian-U.S. collaboration, which has determined the advanced reactor technologies and the advanced technologies of the nuclear fuel cycles to be its spheres of interest. The report gives October'2003 as the date of practical start of this collaboration, which includes joint R&D.

The parties have announced an action plan containing, as a close goal, harmonization of activities within the frameworks of "Generation-IV" International Forum (GIF) and INPRO.

Perspectives for cooperation between USA and Russia in the advanced reactor technology area have also been identified as follows:

— fast liquid metal-cooled reactors (with sodium, lead-bismuth, lead coolant);
— fast and thermal high-temperature helium-cooled reactors;
— reactors fueled with molten salts;
— light-water reactors with supercritical coolant parameters.

The area of advanced nuclear fuel cycle technologies having bilateral cooperation perspectives has been determined as follows:

— development of "dry" fuel reprocessing methods;
— development of closed fuel cycles for thermal and fast reactors;
— development of efficient waste management methods;
— development of advanced fuels using uranium, plutonium and thorium in order to ensure their more efficient use, higher proliferation resistance, and higher environmental safety level for nuclear power facilities.

On the whole, bilateral activities of 2002 – approved at the supreme level – has made it possible to identify areas of mutual interest, bring out unsolved legal and political issues, work out recommendations for potential collaboration in developing the advanced nuclear technologies, and outline a definite action plan.

Community of opinions existing in the USA and in Russia about vital objectives of the advanced nuclear technologies has also been demonstrated. Coincidence of basic principles has been fixed, in spite of several divergences. The parties had to face the need of transition to the practical activity phase.

4. PROBLEMS

The following – though quite short yet – period has clearly demonstrated the "transition problems":

— The governmental experts' report submitted to the leaders of the two countries recognizes the existence of "political issues, which presently restrain nuclear cooperation between Russia and the United States".
— Today this is the U.S. disapproval of Russian participation in the Iranian nuclear power program. This position is not limited to the Bushehr NPP construction criticism – it manifests itself in "excommunicating" some Russian scientific organizations (including one of the country's leading research and design companies, ENTEK) from participation in U.S.-Russian contracts. The 2003 events, unfortunately, show the possible extension of the "disagreement zone".
— Russian nuclear power industry, which naturally focuses its investment resources on new NPP construction, has limited capabilities in the field of advanced nuclear technology R&D. In addition, the trend towards priority financing of a single reactor

concept (lead-cooled reactor) that is unjustified by an objective analysis of future nuclear power structure persisted until recently.

— The U.S. advanced reactor program until recently has been reduced to commercialization of ALWR technology. Fuel cycle research has been considerably limited in 70-80ies, and stopped completely in the beginning of the 90ies, with dismantling of major part of the research infrastructure. The U.S. government's turn to the "new energy policy" and provision of the required assistance to the industry is a quite inertial process.
— Even before the attempt to agree their advanced nuclear technology positions, which took place in 2002 and was successful enough, Russia and the United States have taken different ways in their activities aimed at organizing international collaboration on new nuclear energy systems (IAEA international project INPRO and the international forum "Generation-IV"). Up to now these programs are developing almost independently.

5. PERSPECTIVES

Perspectives of overcoming the transition period difficulties in the U.S.- Russian cooperation in the field of advanced nuclear power technologies clearly manifest themselves in the joint address of 6 U.S. national laboratories to the Secretary of Energy, calling DOE to "implement a comprehensive and integrated plan to further the development and deployment of nuclear energy and the management of nuclear materials".

The principles contained in this letter are also worth citing:

— On the world scale, energy security is closely connected with the national one. Energy supply influences international relations, environment and global well-being. Advanced nuclear systems can provide energy, while considerably mitigating the carbon energy sources' impact on the environment and ensuring important industrial processes (for example, water desalination).
— An integral, wide approach is needed to create a global architecture including all the variety of reactor designs and fuel cycles, as well as reprocessing and waste management, with corresponding safeguards and non-proliferation measures.
— Also, a multilateral governmental cooperation and partnership with the industry, national laboratories and universities is required. USA and Russia, as the founders of the "nuclear era", have a special responsibility for nuclear materials' management and non-proliferation. Advanced approaches to nuclear fuel cycle and already extracted nuclear weapon-grade materials may be used to reduce nuclear proliferation risk.

"We believe", - say the authors, - "the long-term challenges of energy and national security and protection of the environment are closely coupled. To reduce the global nuclear threat while enabling nuclear power to contribute significant amounts of carbon-free electricity, fresh water, and hydrogen, close cooperation and collaboration among all participants is essential".

The "Action Plan" proposed by the national laboratories contains ambitious goals: to achieve 50% of the U.S. electricity production by nuclear sources by 2050, and, also by the mid-century, to ensure 25% of the country's transport fuel production by "nuclear" hydrogen. Achievement of 90% reduction of reactor waste amounts – for which the demonstration of the closed fuel cycle technological system should take place already in 2020 – is also expected by 2050.

A necessary prerequisite for that lies in the rapid growth of governmental investments into nuclear R&D (from 150 million USD today to already 500 million USD in 2008).

It should be noted that the recommendations of the laboratories' directors related to creating a demonstration high-temperature nuclear reactor by 2015, and "selecting one single fast reactor system by 2010, with its further demonstration in 2020", definitely suppose the extension of international cooperation with the participation of Russia, which has a considerable experience in the field of several reactor concepts considered within the frames of GIF program.

Promising U.S.-Russia cooperation perspectives are supported with practical actions taken on the laboratory level. The above-mentioned Sandia-Kurchatov collaboration is continued, with joint commitment of the both laboratories "to assist their countries' governments in determining the ways to solve the problems presently faced by nuclear power cooperation". The parties pledged to perform, in 2003, an analysis, sufficient to illustrate an optimal energy infrastructure including nuclear fuel cycle.

There are hopes that this partnership will be joined by the group of several U.S. national laboratories and other Russian organizations, in order to promote the development of multilateral initiatives realized in collaboration with other nuclear countries.

This position finds support in the U.S. political circles. Below follows the statement made by the U.S. Congressman Curt Weldon at the Moscow Conference, dedicated to the centenary of Igor Kurchatov (January'2003):

"It is appropriate that we look to move to a new level of cooperation in nuclear science that forges a 21st century U.S./Russian alliance that builds on and rededicates our two great nations to the peaceful uses of nuclear energy for the improvement of the quality of life for all human beings on the face of the Earth.

I propose that we create the Kurchatov-Teller Alliance for Peace that brings together in a formal way the Kurchatov Institute and the labs of the Ministry of Atomic Energy with Lawrence Livermore Laboratory, Oak Ridge, Argonne, Los-Alamos and the labs of our Department of Energy for the specific purpose of enhancing the use of nuclear power worldwide while controlling proliferation".

In this connection, the recent statement on science and technology for sustainable development made by G8 leaders in Evian - "We take note of the efforts of those G8 members who will continue to use nuclear energy, to develop more advanced technologies that would be safer, more reliable and more resistant to diversion and proliferation," – practically approves bilateral cooperation of the two nuclear power founders, as a positive factor of the world nuclear development in the long term.

6. ACKNOWLEDGEMENTS

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U.S. DEPARTMENT OF ENERGY INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

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Abstract. In January1997, the President of the United States requested his Committee of Advisors on Science and Technology (PCAST) to review the current national energy research and development (R&D) portfolio, and provide a strategy to ensure that the United States has a program to address the Nation's energy and environmental needs for the next century. In 1999, in response to the PCAST recommendations, DOE established the Nuclear Energy Research Initiative (NERI). Recognizing the importance of a focused program of international cooperation, PCAST issued a June 1999 report entitled *Powerful Partnerships: the Federal Role in International Cooperation on Energy Innovation*. It highlights the need for an international component of the NERI program to promote "bilateral and multilateral research focused on advanced technologies for improving the cost, safety, waste management and proliferation resistance of nuclear fission energy systems. The report further states that the cost of exploring new technological approaches that might deal effectively with the multiple challenges posed by conventional nuclear power are too great for the United States or any other single country to bear, so that a pooling of international resources is needed. In 2001, the U.S. Department of Energy (DOE) established the International Nuclear Energy Research Initiative (I-NERI) to help overcome the principal technical and scientific issues affecting the future use of nuclear energy and to foster global cooperation in nuclear technology.

1. INTRODUCTION

The International Nuclear Energy Research Initiative (I-NERI) was established by the U.S. Department of Energy (DOE) in 2001 as a mechanism for coordinating international research and development (R&D) on next-generation nuclear energy systems known as *Generation IV*. I-NERI was established in response to recommendations of the Presidents' Committee of Advisors on Science and Technology (PCAST) in the Committee's 1999 report entitled *Powerful Parntnerships: The Federal Role in International Cooperation on Energy Innovation*. I-NERI along with the Generation IV Nuclear Energy Systems Initiative, are key elements of the Federal effort to foster global cooperation in development of advanced nuclear energy technology.

The I-NERI program has made substantial progress toward achieving its goals of addressing potential technical and scientific obstacles to the continued global implementation of nuclear energy, establishing of international collaborations for devleopment of advanced nuclear technology, and enhancing the nuclear energy infrastructure.

2. DESCRIPTION OF I-NERI PROGRAM

Over the last thirty years, nuclear power has risen to become the second most important source of electric energy in the United States and at the same time, the most operationally economic. The benefits of nuclear power as a clean, reliable and affordable source of energy are a key to the economic and environmental foundation of the world. The I-NERI vision is to maintain a viable nuclear energy option that will help meet future global energy needs. It's goal is to sponsor innovative investigator-initiated R&D in bilateral cooperation with participating countries to develop advanced technologies to improve cost performatnce, enhance safety, and increase proliferation resistance of future nuclear energy systems in the world. Bilateral I-NERI agreements are normally established under existing or new "umbrella' agreements between the collaborating countries. The U.S. element of I-NERI is managed by the Office of Nuclear Energy Science and Technology who received guidance from the Nuclear Energy Research Advisory Committee (NERAC). A counterpart agency of the collaborating country manages their participation.

In the I-NERI Program, collaborative R&D proposals are solicited simultaneously in the United States and in the participating country. Competitive solicitations issued simultaneously by DOE and by the collaborating foreign agency result in submissions of collaborative researcher-initiated proposals. Eligibility includes Federal laboratories, universities, and industry from the United States and collaborating countries. R&D organizations from these collaborating countries form research teams to develop integrated project proposals. Proposals are formally reviewed and the best potential collaborative projects are selected based on an integrated, peer- review selection process. The specific solicitation workscope and the review, and selection processes are tailored to the terms of each I-NERI agreement.

Peer review panels are selected in each country based on their technical expertise and capabilities in the fields for proposals that they will review. A common set of proposal evaluation criteria, which are established by the Bilateral I-NERI Steering Committee (BINERIC), is used by each country in the peer review process. Separate peer reviews of the collaborative proposals are conducted by the United States and by the collaborating countries to determine the technical and scientific merit of each proposed project. NE receives a rank order list of the proposals from the peer review based on technical and scientific merit. A Federal programmatic review of the proposal is performed to ensure that proposed projects comply with Department policy and programmatic requirements including clearly defined scope, deliverables, end products, and planned cost and schedule. Analogous reviews are conducted simultaneously by the collaborating agency based on the rank order listing provided by their peer review panel. Final award selections of high merit, mutually beneficial proposals are made by the BINERIC in executive session via joint evaluation of the respective peer review results and recommendations.

3. INTERNATIONAL AGREEMENTS

To date, four I-NERI collaborative agreements have been established; the first between DOE and the Commissariat a l'Inergie Atomique of France, the second between DOE and the Republic of Korea Ministry of Science and Technology, the third with the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency, and the fourth with the European Commission (EC) European Atomic Energy Community (EURATOM). Since the program inception, five projects with France, eleven with the Republic of Korea, and one with the Nuclear Energy Agency have been initiated. Discussions on collaboration are ongoing with Brazil, Canada, Japan, the Republic of South Africa, and the United Kingdom.

4. UNITED STATES/REPUBLIC OF KOREA COLLABORATION

The United States/Republic of Korea (ROK) collaboration focuses on advanced technologies for improving the cost, safety, and proliferation resistance of nuclear energy systems. Eleven projects have been awarded and include projects in the areas of advanced instrumentation, controls and diagnostics; advanced light water reactor technology; next generation reactor and fuel cycle technology; innovative nuclear plant design, manufacturing, construction, operation, and maintenance technologies and advanced nuclear fuels and materials.

Solicitation for the U.S./ROK I-NERI collaboration was held in July 2001 and July 2002. Six projects were awarded in December 2001 and five projects awarded in December 2002. The R&D topical areas selected by the U.S./ROK Bilateral I-NERI Committee (BINERIC) for these procurements included the following:

— Advanced instrumentation, controls and diagnostics
— Advanced light water reactor technology
— Next generation reactor and fuel cycle technology (including non-proliferation and safety)
— Innovative nuclear plant design, manufacturing, construction, operation, and maintenance technologies (including instrumentation, controls, and robotics)
— Advanced nuclear fuels and materials

5. UNITED STATES/FRANCE COLLABORATION

The U.S./France collaboration focuses on the development of Generation IV advanced nuclear system technologies that will enable the United States and France to move forward with leading edge R&D that can benefit the range of anticipated future reactor and fuel cycle designs. Five projects have been awarded and include the design of a Gas Fast neutron Reactor (GFR); nano-composited steels capable of meeting the high operating temperatures of the Gen IV reactors; computational modeling for gas reactor coated particle fuels; computational modeling for improving neutronic predictions of advanced nuclear fuels; and hydrogen production from nuclear energy.

The U.S./France collaboration was the first I-NERI agreement to be implemented. The competition for project awards was limited to DOE and CEA laboratories in recognition of limited budgets available and desire of the parties to facilitate timely initiation of the program. R&D topical areas selected in FY 2001 by the U.S./France Bilateral I-NERI Committee (BINERIC) for the initial competition were as follow:

— Advanced Light Water-Cooled Reactors

— Advanced Gas-Cooled Reactors
  - Advanced Gas-Cooled Reactor Concepts
  - Fuel Development
  - High Temperature Systems Technology
  - Mechanistic Behavior Model for TRISO Fuel Particles
— Advanced Fuel and Materials Development
  - Nano-Composited Steels
— Radiation Damage Simulation
— Advanced Fuel Cycle Chemistry

Three projects were awarded to teams of DOE and CEA federal laboratories in FY 2001 and a fourth was awarded in FY 2002. A fifth project was added to the DOE/CEA collaboration at the end of FY 2002 in the area of *Hydrogen Production using Nuclear Energy*.

6. UNITED STATES/OECD NUCLEAR ENERGY AGENCY COLLABORATION

The United States has teamed with the Nuclear Energy Agency (NEA) in collaborative R&D to conduct reactor materials experiments and associated analysis. The U.S. funding team consists of the Nuclear Regulatory Commission, the Electric Power Research Institute, and DOE. A project has been awarded on the *Melt Coolability and Concrete Interaction (MCCI)* to achieve improved accident management guidelines for existing plants and also develop better containment designs for future plants.

The Melt Coolability and Concrete Interaction (MCCI) project is a collaboration between Argonne National Laboratory, the U.S. Nuclear Regulatory Commission (NRC) and a consortium of international participants represented by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). All tests and associated analysis are carried out at Argonne National Laboratory. The project was authorized in March of 2002.

The focus of work on this project is to assess the ability of water to quench and thermally stabilize a molten core-concrete interaction, which is an important outstanding question in the area of reactor safety.

7. CONCLUSION

The Office of Nuclear Energy, Science and Technology is coordinating a wide-ranging discussion among governments, industry, and the research community worldwide on the development of next generation reactor and fuel cycle systems. I-NERI is DOE's key collaboration mechanism for conducting R&D with the international research community. Additional information regarding DOE's I-NERI program, Generation IV Nuclear Energy Systems Initiative, and Advanced Fuel Cycle Initiative is available at www.nuclear.doe.gov.

INNOVATIVE NUCLEAR SYSTEMS AND RELATED R&D PROGRAMMES

(SESSION 6)

Chairpersons

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IAEA

D. NICHOLLS
South Africa

INNOVATIVE REACTOR TECHNOLOGIES - ENABLING SUCCESS

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Abstract. Many innovative reactors are being discussed, offering advantages in economics, sustainability, environmental impact, versatility and efficiency. To be successful, however, innovative reactors must meet the requirements for a successful build project. This requires achieving the mixture of innovation and proveness required to meet the first-of-a-kind hurdle. Based on the successful CANDU 6 reactor, a design still being built today, the ACR adds specific innovations in key areas chosen to achieve a balanced design. Capital cost has been significantly reduced by optimising the reactor-core design and simplifying systems. Key changes in this area include a move from a heavy water coolant to a light water coolant, and the adoption of SEU fuel. Construction times have also been reduced by using a modular design that takes advantage of modern construction techniques. Operating performance has been enhanced through improvements in system materials, equipment layout and component specifications. In parallel with these priorities, design adaptations have been applied so as to increase safety margins and defence-in-depth, again adding to the confidence in ACR licensability. The ACR development plan includes early review by regulators to reduce licensing risk, with international regulatory review having commenced. Overall, this places the ACR in a good position to meet the first-of-a-kind challenge, a necessary condition to enabling the success of an innovative reactor. AECL sees a logical evolution from the ACR, via increasing temperature and pressure capability, to the SCWR (Supercritical Water Reactor). AECL's CANDU-X program is already looking at designs for this concept. Inherent features of both ACR and the fuel channel SCWR lend themselves to different fuel cycles for the future. One of the prominent characteristics of the heavy-water moderated fuel channel reactor approach is the high potential for innovation. The evolutionary path allows innovation in practical steps, yet allows far reaching improvements to be ultimately achieved.

1. INTRODUCTION

Many innovative reactors are being discussed in forums such as the present IAEA symposium, targeting advantages in economics, sustainability, environmental impact, versatility and efficiency. These innovative approaches are being developed with a view to long-term sustainability through linkages to fuel cycle development. To be successful, innovative reactors must combine short-term and long-term benefits together. In the short-term, innovative reactors must meet the requirements for a successful build project. This requires achieving the mixture of innovation and proveness required to meet the first-of-a-kind bundle, so that initial build projects can proceed.

AECL has been pursuing innovation in reactor design using an evolutionary approach. This enables manageable, short-term innovative steps, but retains a long-term direction that can extend the evolution of design in far-reaching ways. The Advanced CANDU Reactor, or ACR design is the logical next step in the CANDU fuel-channel reactor design process, and achieves major improvements in economics while expanding safety margins. The ACR design is evolved from AECL's current CANDU 6 design with a specific series of enabling technologies which have been developed at AECL and are now being proof-tested.

The technology applied in the ACR is also the start of the series of steps that represent the long-term development path for CANDU fuel channel reactors. The ACR, as a light-water cooled fuel channel reactor, shares much in common with other light water reactors. In the same way the further evolution of the CANDU line fits in with the long-term development directions identified in initiatives such as the Generation IV International Forum (GIF) and INPRO. Specifically, AECL sees the product evolution from the ACR, via increasing pressure and temperature, to the Supercritical Water-costed Reactor (SCWR), one of the concepts identified by GIF for further development. AECL's "CANDU-X" program, looking 25 years into the future, has identified the heavy-water-moderated fuel channel SCWR as the natural evolution of the CANDU family. This would have a number of natural advantages as a readily developed, economical SCWR with a high level of safety assurance.

Inherent features of both ACR and the fuel-channel SCWR lend themselves to the adaptation to different fuel cycles of the future. Efficient use of neutrons, on-power refuelling, and simple, adaptable fuel bundle designs all enable fuel cycle benefits in the future.

One of the prominent characteristics of the heavy-water moderated fuel channel reactor approach is the high potential for innovation. The evolutionary path allows innovation in manageable steps, yet enables far-reaching improvements to be ultimately achieved.

2. BACKGROUND TO DEVELOPMENT STRATEGY

AECL has focused on the CANDU fuel channel reactor design, and its development potential over many decades, and continues to see this as a centrepiece of nuclear development. As part of the pressurized water reactor family, CANDU's share many characteristics with other light water reactors, while retaining a set of distinctive features:

— High pressure water coolant in individual fuel channels, with low-pressure, low temperature moderator
— Horizontal fuel channel design with on-power refuelling
— Simple, easily-fabricated fuel bundle design
— Use of heavy water to improve neutron efficiency.

The original impetus for these features was to enable a practical nuclear energy system to be developed in Canada in the decades following the Second World War, when the available industrial infrastructure was limited. The result was a nuclear reactor design which eliminated the need for a large high-pressure reactor vessel, which enabled all reactor core instrumentation and control elements to be located in the low pressure, low temperature moderator, and which enabled very low uranium requirements (and, with heavy water coolant, eliminated the need for uranium enrichment).

The subsequent decades leading up to today have seen many changes: in the global industrial infrastructure; in the understanding of uranium and other fuel resources; in safety and environmental protection expectations. With these changes, the basic principles underlying the CANDU approach remain valid, while design adaptations can readily align the CANDU system with current and future economic realities.

Since the original CANDU reactors were first built, the advent of deregulated, competitive energy markets has strongly emphasized the importance of low capital cost and short construction time. At the same time, increased availability of uranium resources and enrichment technology has decreased the pressure on fuel costs, while strategic, security-of-supply considerations stress the need for fuel adaptability during the lifetime of the next generation of reactors.

The ACR design innovations are chosen to respond to this evolution of energy markets, while retaining the proven features of the CANDU line.

3. DESCRIPTION OF ACR PRODUCT

The ACR-700 design is an evolutionary development of familiar CANDU technology, adding a carefully chosen set of innovations to the major improvements in economics, operations and safety margins. With a gross electrical output of approximately 728 MWe, the ACR follows the same size range as AECL's standard CANDU 6 design, allowing much of the extensive experience base in CANDU 6 design, construction and operation to be utilized.

The ACR-700 design is rooted in the proven principles and characteristics of the CANDU system, and uses standard features of CANDU pressure tube technology built up over many decades of operation:

- Modular horizontal fuel channel,
- Available simple, economical fuel bundle design,
- Separate cool, low-pressure heavy water moderator with back-up heat sink capability,
- On-line/at power fuelling,
- Fuel cycle flexibility with high neutron efficiency,
- Passive moderator/shield tank heat sinks surrounding the pressure tube core,
- Two robust, quick acting, passive shutdown systems.

The following key features, derived from the enabling technologies, are incorporated into the design concept of the ACR

- Slightly enriched uranium fuel (nominally 2% U-235), contained in CANFLEX bundles to achieve burn-up (approximately 20 MWd/kgHM) and with further increases as operational experience increases,
- Light water replacing heavy water as the reactor coolant,
- More compact core design with reduced lattice pitch, reducing heavy water inventory and providing a highly stable core neutron flux,
- Enhanced safety margins, due to optimized power profile and void reactivity,
- Higher coolant system and steam supply pressure and temperature resulting in an improved overall turbine cycle efficiency,
- Reduced emissions, due to radiolysis of heavy water,
- Improved performance through advanced operational and maintenance information systems, and improvements to project engineering, manufacturing and construction technologies.

A simple diagram of the ACR-700 design is shown in Figure 1, and main design parameters are given in Table I.

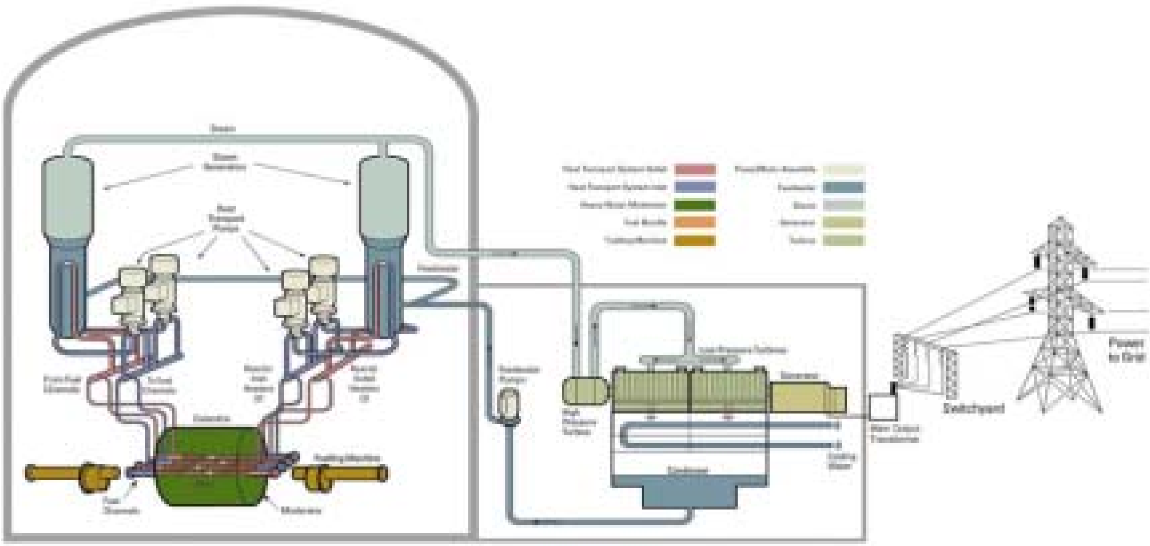

FIG. 1. CADDS diagram of ACR coolant system.

Table I. ACR main design parameters

Reactor thermal output:	1983 MW (th)
Nominal plant electrical output:	728 Mwe (gross)
Reactor coolant system pressure	12MPa (at reactor outlet header)
Reactor coolant system inlet temperature:	278.5º C
Reactor coolant system outlet temperature:	325º C
Nominal boiler steam pressure:	6.5MPa (a)
Nominal boiler feedwater temperature:	218º C
Number of fuel channels:	284
Number of fuel bundles per fuel channel:	12
Fuel design:	43-element CANFLEX fuels

The ACR has been designed from the initial conceptual stage with both constructability in mind.

The ACR is designed for highly modular construction. The entire reactor building internal equipment is assembled as a series of 105 modules, to be installed in “open-top” approach using a very heavy lift crane. The building arrangement allows the longest lead-time equipment to be installed at the latest date. This approach, coupled with the comprehensive use of a suite of electronic engineering and project tools, focussed on 3-D CADDS plant models, allows rapid construction and overall project schedules.

With a construction period (first concrete to start of main commissioning) of 36 months, the total project schedule for a first-of-a-kind ACR is 60 months, from contract effective date to in-service. For replica units (so called “n’th units”) this is targeted to reduce to 48 months.

The ACR is also designed for ease of operation. The improved operation information systems, simplified system design, and layout for easy access, mean that the operating cost is reduced. By taking advantage of on-power refuelling, the plant is designed for a three-year interval between outages, and a target planned outage duration of 21 days.

4. DESCRIPTION OF FUEL CHANNEL SCWR

4.1. CANDU X

Research underway on the advanced CANDU studies new, innovative, reactor concepts with the aim of significant cost reduction and resource sustainability through improved thermodynamic efficiency and plant simplification. The so-called CANDU-X concept retains the key elements of the current CANDU designs, including heavy-water moderator that provides a passive heat sink and horizontal pressure tubes. Improvement in thermodynamic efficiency is sought via substantial increases in both pressure and temperature of the reactor coolant. Following on from the new Next Generation (NG) CANDU, which is ready for markets in 2005 and beyond, the reactor coolant is chosen to be light water but at supercritical operating conditions. Two different temperature regimes are being studied, Mark 1 and Mark 2, based respectively on the continued use of zirconium or on stainless-steel-based fuel cladding. Three distinct cycle options have been proposed for Mark 1: the High Pressure Steam Generator (HPSS) cycle, the Dual cycle, and the Direct cycle. For Mark 2, the focus is on an extremely simple direct cycle.

Supercritical water (SCW) as reactor coolant becomes feasible with development of the high-efficiency thermallyinsulated fuel channel. The use of compact core lattices allows replacement of heavy water coolant by light water, and becomes viable using a new bore seal for channel closure. By reducing the diameter of end fittings, the channel-to-channel lattice pitch can be adjusted to achieve zero or negative coolant void reactivity, and also fine-tuned with slight shifts in fuel enrichment.

We believe the pressure tube concept allows for great flexibility in the design of an SCWR, as the density, power and flux profiles can be optimized using the standard CANDU interlaced flow paths. Moreover, the use of SCW is not new, since some existing coal plants already use SC turbines at power ratings in excess of 800 MW(e) and attain overall cycle thermal efficiencies of > 40%. Supercritical coolant pressures permit large changes in enthalpy with small changes in temperature without encountering the two-phase region with its critical-heat-flux limitations.

The current CANDU X concepts are divided into two groups:

— Mark 1: uses zirconium based fuel cladding for high neutron-efficiency. Avoidance of excessive corrosion places a limit on coolant core-outlet temperature of some 420O C. Higher temperatures might be possible if the cladding surface is successfully treated with a thin corrosion-resistant layer.
— Mark 2: trades some neutron efficiency for thermodynamic efficiency: stainless-steel-clad fuel permits coolant outlet temperatures as high as 625O C, consistent with attaining the greatest efficiencies at the highest inlet temperatures of modern SC turbines.

To help guide the development of CANDU technology for deployment in the 2025-2030 timeframe, AECL maintains its CANDU X program. The CANDU X is an advanced concept targeted for deployment in the 2025 timeframe. The CANDU X is heavy-water moderated CANDU that is cooled by supercritical water (SCW). The use of SCW is not a new concept in power generation, having been used in fossil-fired plants for over 30 years. It is new, however, for nuclear plants, where a number of development challenges have to be overcome. Fortunately, CANDU reactors are very amenable to using SCW compared to LWRs for two reasons:

— Because the coolant and moderator in a CANDU are separate, the reactor is relatively insensitive to the large changes in coolant density that can occur across the core of a SCW cooled reactor.
— Designing a reactor core to handle the increased pressures and temperatures associated with SCW cooling is much simpler with a fuel-channel reactor than a pressure-vessel reactor.

In addition to these technical advantages, some potential safety advantages evolve from a move to SCW in a CANDU-type reactor.

5. FUEL CYCLE OPTIONS

At present, the primary nuclear fuel cycle worldwide is a once-through uranium cycle. At today’s uranium and enrichment prices, the front-end and operating costs of this fuel cycle represent a small fraction (typically less than 10%) of total energy cost, and are therefore very competitive. Based on a once-through cycle, substantial energy content remains in the used uranium fuel. However, competitiveness of recycle options, led by MOX, remains uncertain, based on high forecast prices for reprocessing and continued low uranium and enrichment prices forecast into the future.

This means that the next generation of innovative plants will start their life with once-through uranium as the most likely fuel cycle of choice. However, given plant lifetimes of 60 years, it is likely that fuel cycle economics will change during the life of these plants. Therefore the ability to adapt to different fuel options is an important design attribute to consider.

In keeping with the CANDU tradition of neutron efficiency, the ACR is highly adaptable to a range of fuel cycles. The ACR is a very effective user of MOX fuel. The ACR can burn a 100% MOX core, without change to permanent reactor equipment, and can transition from a conventional SEU core to a MOX core without shutdown, using on-power refuelling. As ACR fuel burnups increase, the cost of MOX fuelling decreases. MOX fuel is relatively straightforward to manufacture in ACR fuel bundles.

Further, the ACR is a practical user of thorium-based fuels. A number of viable options for thorium fuelling have been identified, using the advantages of on-power fuelling and simple bundle design.

Thorium requires a core that includes fissile driver fuel. The use of on-power refuelling enables such a mixed core to be readily adjusted, and allows thorium fuel bundles to be shuffled so that they can be used as is, without the need for reprocessing to extract U-233 material.

The fuel channel SCWR core design will retain these attributes, since it continues with the common CANDU-based features. In addition, by increasing thermal efficiency, a further benefit in electricity product per unit of burnup is obtained.

6. FUEL CHANNEL REACTOR DEVELOPMENT POTENTIAL

Cooperative international activities such as GIF and INPRO are exploring long-term nuclear development options and identifying approaches to set development priorities. Typically, development review activities look at desirable end-points for global systems or for technologies. AECL has been part of this activity in support of Canada's role in the global nuclear community. As an organization responsible to establish a business case for reactor development, AECL has also reviewed the practicality of the development path – an important consideration in prioritizing development strategies. In this regard, the heavy-water moderated fuel channel design approach offers significant advantages, as follows:

6.1. Economical development

The development program for the ACR and onward to the fuel channel SCWR is a relatively economical activity:

- The evolutionary approach means that for each new design, a limited set of new technologies or components need be demonstrated. All the features of the SCWR do not need to be developed at once; rather, the ACR will form the originating technology framework, supported by development of materials for use at steadily increasing temperatures until the SCWR final conditions are reached.
- The modular fuel channel design approach means that the basic technology unit to be developed -- the fuel channel itself -- is relatively simple and small in scale. Testing of a single fuel channel represents a full core in scope. Similarly, the small, simple fuel bundle design can be readily developed and test irradiated at low cost.
- As part of the LWR family, ACR and SCWR development builds on the extensive existing investment in water reactor technology in Canada and around the world.

6.2. Economical deployment

The characteristics of fuel channel reactors lend themselves to economical build of prototype and first-of-a-kind units. Tooling and manufacturing development costs for modular components such as fuel channels are more readily absorbed in production runs of multiple components.

Further, by staging development in a small series of manageable, evolutionary steps, each successive design stage can use a large amount of design, licensing, equipment and construction technology from the previous stage, minimizing first costs. The ACR-700, developed as an evolutionary step from AECL's current CANDU 6 design, keeps first-of-a-kind costs low enough to allow economically competitive deployment right from the first unit.

6.3. Low development risk

The evolutionary development approach reduces risk as well as cost. By moving toward the SCWR in a series of steps, the innovations in each step represent a modest technology risk which can be assessed through testing and demonstration and then performance in first units.

The fuel channel approach also has inherent risk reduction advantages. Individual fuel channels have been, and will continue to be designed for replacement if necessary; other core components in the low-pressure, low temperature moderator environment, see modest duty which is easily replicated in tests, and are also readily designed for adjustment or replacement. Finally, fuel designs, in particular alternate fuels, are easily demonstrated in a power reactor environment due to on-power refuelling and the simple CANDU fuel design. A single channel, or set of demonstration channels, can be use to demonstrate successive adaptations in fuel design or increases in burnup.

6.4. Scope of development potential

Relative to other current reactor types, the heavy water moderator fuel-channel reactor family has an extremely broad scope for development. For most of the past, development has been focused on refining and improving the familiar natural-uranium fuelled CANDU line. While extensive development of enriched uranium and other alternate CANDU fuels has been completed, ACR represents the first optimization of the CANDU concept to take advantage of slightly enriched fuel. The result is a significant opportunity for economic benefits, broadening the safety envelope and system simplification. Once the ACR is established, further improvements can be introduced, such as increased thermodynamic efficiency via increased steam pressures and reduction in fuelling overall costs via increased burnup.

6.5. Flexibility

The fuel channel concept has characteristics naturally suited to fuel cycle flexibility, as noted above. Innovative designs are tailored to ensure this flexibility is maintained or extended. In addition, there is great flexibility in deployment. Manufacturing of CANDU fuel and plant components can be readily introduced to host countries, because of the simpler, smaller-scale nature of the key components.

Finally, the modular nature of the fuel channel approach means that the same stage of technology can be deployed in different power levels. For example, AECL has established the ACR technology in both a reference ACR-700 (700MWe) design and an alternate ACR-1000 (1000MWe) design, each with the same system and component technology and the same licensing basis. The technology can be readily adapted to larger or smaller power outputs, depending on market demand.

6.6. Synergies with other technologies

Looking into the long-term, and at an evolving future, more than one innovative nuclear technology is likely to be needed. In particular, in order to balance the demands of economical energy with long-term conservation of fuel resources, a blend of "breeder" and "burner" technologies would be needed. As a fuel-efficient burner technology, heavy-water moderated fuel channel reactors represent a good complement to breeder technologies as they are developed. As part of the water-reactor design continuum, fuel channel reactors can make use of, and contribute to, a very wide R and D and experience base.

7. CONCLUSIONS

AECL has established a development path forward for the CANDU family of fuel channel reactors. The next step, the Advanced CANDU Reactor (ACR) design, offers significant improvements in economics and safety case, and retains the traditional CANDU flexibility in adapting to alternate fuels. For the longer term, the fuel channel reactor type is an attractive option for supercritical water reactor (SCWR) applications. Development of fuel channel reactors along this path offers some important benefits in: economical development and deployment; low development risk; scope of development potential; flexibility; and in synergies with other technologies.

OVERVIEW OF THE CEA PROGRAMME ON HIGH TEMPERATURE GAS COOLED NUCLEAR ENERGY SYSTEMS: THE “GAS TECHNOLOGY PATH”

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Abstract. The so called “Gas Technology Path” is a practical guideline for a sequenced development of high temperature gas cooled reactors. It has been selected as main focus for CEA R&D plans on future nuclear energy systems, as a result of an early technology roadmap performed at the end of 2000. The selection of this research objectives originates both from the significance of fast neutrons and high temperature for nuclear energy to meet the needs anticipated beyond 2020/2030, and from the significant common R&D pathway that supports both medium term industrial projects and more advanced versions of gas cooled reactors such as a Very High Temperature Reactor (VHTR) for massive production of hydrogen, and a Gas Fast Reactor (GFR) with a closed fuel cycle for sustainable nuclear power. The first step of the “Gas Technology Path” aims to support the development of a modular HTR likely to meet international market needs around 2020. To this end, the CEA is in the process of developing up-dated HTR technology R&D means to support Framatome-ANP’s initiative to diversify its portfolio of reactors beyond large evolutionary Light Water Reactors (EPR and SWR-1000). The second step is a Very High Temperature Reactor (> 950 °C) to efficiently produce hydrogen though thermo-chemical water splitting or to generate electricity with an efficiency above 50%, as well as to offer opportunities of very high temperature process heat applications such as steel or aluminium production. The third step of the Path is a Gas Fast Reactor that features a fast-spectrum helium-cooled reactor and closed fuel cycle, with a direct-cycle helium turbine for electricity production. The GFR’s fast spectrum makes it possible to extract most of the energy content of the available fissile and fertile materials as opposed to thermal spectrum reactors which typically burn 1 % of natural Uranium only. Furthermore, through the combination of a fast neutron spectrum and full recycle of actinides, GFRs minimize the production of long-lived radioactive waste. The GFR concept targets an integrated, on-site, spent-fuel treatment and re-fabrication plant. The VHTR and the GFR were confirmed in October 2002 as being among the six Generation IV systems selected for further development in the frame of a multilateral collaboration effort. The paper succinctly presents the R&D program launched in 2001 by the CEA with industrial partners on the “Gas Technology Path”, which is destined to become the contribution of France to the development of the VHTR and the GFR within the next phase of the Generation IV Forum.

1. INTRODUCTION: THE INTERNATIONAL AND THE NATIONAL CONTEXT

Research and Development work on future nuclear energy systems in France applies to innovative reactor, fuel and the fuel cycle technologies, beyond the options developed for current industrial projects. These researches are organized in two major strategic axes:

(i) Innovations for pressurized water reactors ;
(ii) 4th generation nuclear energy systems with a main focus on the development of gas-cooled systems, and more prospective activities on other candidate technologies:
 — up-dating the vision of sodium cooled fast neutron systems (given the extended expertise acquired on Phenix and SuperPhenix);

— assessing the feasibility and potential performances of supercritical water cooled systems, and molten salt reactors.

Companion studies on scenarios of electronuclear development and on techno-economic evaluations, participates in the orientation of these researches for innovations and insures the global coherence.

The researches for innovations for light water reactors are essentially led in a national frame, in cooperation with the industrial partners of the CEA. They are focused first and foremost on the technological headways on fuels allowing progress in economy, safety and management of the plutonium (and potentially that of some minor actinides) in the French fleet of reactors. At a lesser level, these researches participate in the reflections of Framatome-ANP on its reactors portfolio and in particular on the segment of PWR market in the range of 300 to 600 MWe.

Beyond possible progresses in PWRs, the CEA led an internal reflection at the end of 2000 on the most promising technical options for systems (i.e. reactors & fuel cycles) of the 4th generation allowing a sustainable energy development, and likely to meet the following criteria, widely shared internationally:

(i) A strengthened economic competitiveness (notably by the reduction of capital costs);
(ii) A increased security as well as an increased safety, during the plant operation, in accidental situations, and regarding risks of proliferation;
(iii) A reduced impact on the environment:
  — by a flexible and effective use of nuclear fuel and available resources,
  — by a considerable reduction of the long-lived radioactive elements in the ultimate waste;
  — A capacity for other applications than the production of electricity, such as the production of hydrogen or the desalination of the sea water.

Given the significance of fast neutrons systems, closed fuel cycles and high temperature to meet these goals, the CEA developed a specific interest in a consistent set of gas cooled systems supported by a significant common R&D pathway and offering opportunities for a sequenced development in three stages: a modular reactor with high temperature (850 °C) for the international market around 2020, an up-graded version to very high temperature (> 950 °C) for the massive production of hydrogen, and a system with fast neutrons with full recycling as a vision of sustainable nuclear technology.

The assets of this set of gas cooled nuclear systems have been confirmed by the Technology Roadmap of the Generation IV International Forum in 2001-2002, which led to select the VHTR and the GFR among the six Generation IV systems to be developed internationally in the next phase of the Forum. A budget of 30 MEuros is being allotted to this program in 2003, with a forecast to about double the effort by 2012.

The activity of the CEA on other 4th generation systems aims to share the past expertise acquired in France on sodium cooled reactors in the Generation IV Sodium Fast Reactor (SFR), as well as to further assess the viability and the potentialities of more prospective Generation IV systems such as the Supercritical Water cooled Reactor (SCWR) and the Molten Salt reactor (MSR).

2. HIGH TEMPERATURE GAS COOLED REACTORS: A PATHWAY OF PRIVILEGED DEVELOPMENT

The use of helium allows reaching high conversion efficiencies, while enabling to make use of gas turbines, which made considerable progress and constitute today the most efficient conversion technology.

The French Atomic Energy Commission (CEA) acquired on these reactors an important experience: at first on the technology UNGG (uranium natural graphite gas) in the 60s, then on HTGRs in the 70s and 80s as a result of contributions to the European HTR program and to an active collaboration with General Atomics. The CEA then acquired a wide experience in the fields of particle fuel, systems design and technologies (materials…), as well as in reactors physics. This background expertise is re-invested today in the sequenced development of the three envisioned gas cooled systems.

Gas cooled reactors benefit from a number of attractive features:

— *In terms of security and safety:* the conductive, refractory and highly confining fuel form allows for a very robust behaviour in accidental situations;
— *In terms of economic competitiveness:* a high potential is afforded by a high energy conversion efficiency, a simple design of the primary system (with direct conversion), a probable short duration of construction, that should result in a reduced investment. Additional benefits are also expected from a modular design in the range of 300 MWe offering extended passive safety features and correspondingly simple safety systems.

The main steps of the envisioned gas cooled systems sequenced development path are the following (Fig. 1):

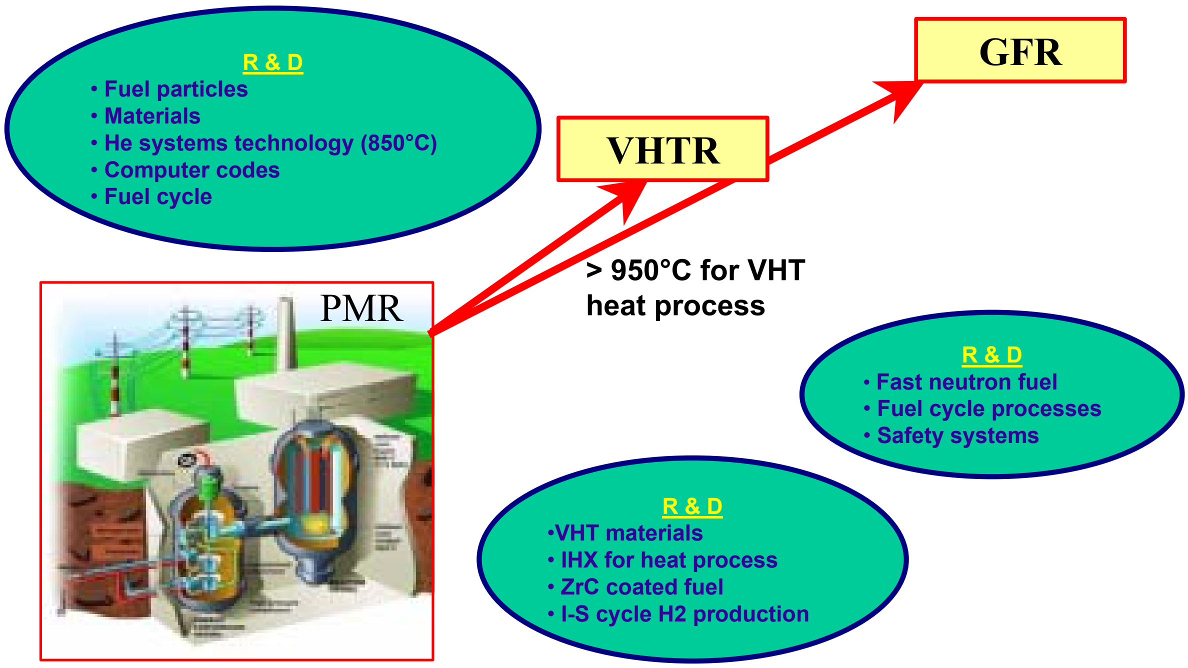

FIG. 1. The "Gas Technology Path" for a sequenced development of high temperature gas cooled reactors.

— By 2015-2020, the development of a modular reactor of 300 megawatts cooled by high temperature helium (850 °C), including the realization of an industrial prototype by 2015. The range of power (300 MWe) presents a specific interest for regions with electricity networks of limited capacity, as well as to equip multi-gigawatt power plants with multiple reactor modules. This may be of specific interest for the deployment of the first units of nuclear production in countries with limited industrial infrastructure.
— The domain of the very high temperatures (greater than 950°C with the VHTR) opens a field of applications beyond the production of electricity such as hydrogen generation to produce synthetic hydro-carbide fuels to replace oil in a few decades; furthermore, co-generation contributes to improving the competitiveness of nuclear power. For these reasons, CEA launched an active R&D program on thermo-chemical water splitting and high temperature electrolysis as most promising processes to produce hydrogen by nuclear power, in collaboration with the Jaeri in Japan and with Sandia national Laboratory and General Atomics in the United States.
— Besides, a technical and economic evaluation of seawater desalination processes likely to be coupled with high temperature reactors was conducted within the framework of the European project Eurodesal and leads to current collaboration with Tunisia and Morocco under the aegis of the International Agency of the Atomic Energy (IAEA).
— Beyond the development of a modular HTR and that of the VHTR, the investigation of a gas cooled fast neutron system (GFR) with innovative refractory fuel and integral recycling of transuranic elements has been launched to re-assess by 2030 the potential of gas as an alternative to liquid metals to cool fast neutron cores. This most advanced edge of the “Gas technology Path” is a vision of a sustainable nuclear system making an efficient use of fertile and fissile nuclear fuels, while minimizing the production of long-lived radioactive waste (to the point where the radio-toxicity of the ultimate waste reaches in a few hundred years the same level as that of the uranium ore used to manufacture the fuel.

2.1. Development of a consistent portfolio of gas cooled systems

The selection of a set of gas cooled systems as guideline for the developments or future nuclear energy systems is simultaneously motivated by the support to the strategy of Framatome-ANP to widen the range of commercialised reactors through the development of a high temperature gas cooled reactor of 300 MWe, and the interest of the CEA - confirmed by the Forum Generation IV - to study the possible adaptations of this system either towards very high temperatures for the production of hydrogen, or towards fast neutrons to attain essential features of sustainable nuclear systems such as breeding fertile fuels and burning all actinides.

These objectives underlie a re-acquisition of HTR R&D including adapted and validated computational tools for gas cooled reactor and system design studies, important developments of high temperature materials and components, and innovations in fuel and fuel cycle processes.

3. PARTICIPATION IN THE DEVELOPMENT OF A MODULAR HIGH TEMPERATURE REACTOR LIKELY TO MEET INTERNATIONAL MARKET NEEDS BY 2020

The current R&D program led in partnership with Framatome-ANP and EDF to support the development of a modular high temperature reactor likely to meet international market needs by 2020 includes the main following items:

(i) Core, fuel and cycle
(ii) There-manufacture of standard TRISO fuel particles (2004), with first irradiation tests within the Osiris reactor in the year 2005 and the objective of a validation by 2010 for an industrial use;
(iii) The development of innovative technologies for the management of the waste generated by the operation of HTR reactors (2006) with the objective of a validation before 2012 for industrial applications
(iv) Design and safety
  — The exploitation of the experience gained on the HTR technology through the organization of the CEA documentary base from the years 1970s and 1980s and sharing the European experience through the network HTR-TN set up within the framework of the 5th European R&D Framework Program;
  — The development of a validated platform of computational tools and procedures for the design studies of gas cooled reactors;
(v) Technology and components
  — The selection and the validation before 2006 of the best materials for a technologically updated HTR, both for graphite structures and for the steels used for the pressure vessel (9Cr-1Mo), for the internal structures, for the gas piping and systems at very high temperature (Haynes 230, Hastelloy XR, Inconel 617, Alloy 800);
  — The deployment from 2003 to 2006 of a set of helium benches and test loops, to experiment the important technological aspects of high temperature helium systems (tightness, tribology, thermal barriers, helium purification); a test loop for components in the range of 1 MW is planned in 2006 (Helite); a system loop in the range of 20 MW (Hello) is envisaged in an international frame around 2009 for tests of coupled big components (heat exchangers, turbine) and for simulating normal or accidental operating conditions;
  — The assessment and the development of hydrogen production processes including the iodine/sulfur thermo-chemical cycle and the high temperature electrolysis.

4. PARTICIPATION IN THE DEVELOPMENT OF THE VHTR SYSTEM WITHIN THE FRAMEWORK OF THE GENERATION IV INTERNATIONAL FORUM

The CEA is a partner of the United States, Japan and other member countries of the Generation IV Forum for the development of the Very High Temperature Reactor (VHTR) system dedicated to massive production of hydrogen and co-generation of electricity. The R&D plan aims to resolve viability issues by 2010 and to confirm the performances in 2015. As partly relying on a significant common R&D pathway, the VHTR will benefit from the earlier development of a modular high temperature reactor as presented above.

The main axes of R&D for the VHTR are the production of hydrogen by the iodine/sulfur thermo-chemical cycle, the technology of very high temperature materials and components (1000°C) required by this process and the intermediate heat exchanger, and the development of fuels offering increased margins for operation at very high temperature. The contributions of the CEA in these fields will be the following:

(i) Core, fuel and fuel cycle
- Development of fuels optimised for the VHTR with the objective of a validation by 2013 for an industrial application: fuel particle with ZrC coating offering a margin of 200°C in comparison with SiC coated TRISO fuel; contribution to the development of fuel particles with UCO kernel to increase the burn-up limits;
- Continuation, beyond the objectives retained for the medium term modular project, of the R&D effort to develop a process for the treatment of spent fuel particles and to develop a technology for the management of the waste generated by gas cooled reactors, with the objective of a validation before 2012 for industrial applications;

(ii) Design and safety
- Support to the design studies of the VHTR system, both in terms of supply of validated procedures of calculation and direct participation in the studies (reference calculations, accidents analyses), to resolve the uncertainties on viability issues (2010) and to confirm the expected performances, in particular as regards the core outlet helium temperature (2015); specific contributions are expected to the studies of coupling modes of the reactor to the hydrogen production processes by thermo-chemical cycle or high temperature electrolysis;
- Participation in the assessment and development of the hydrogen generation processes with the objective to compare around 2006 the performances, the co-generation modes and the coupling features of the iodine/sufur cycle process with those of the high temperature electrolysis. For the iodine/sulfur cycle, the program of the CEA will benefit from an active cooperation with Japan (JAERI) and with the United States (action I-NERI launched in 2002) including the following steps: laboratory scale analytical experiments on the Bunsen reaction (2002-2005), participation in the operation of an experimental loop (100 l/h) (2006-2008), demonstration of production of 150 m^3/hour on a pilot experiment with a hest source of about 1 MW (2008-2012), participation in Jaeri's nuclear hydrogen production experiments with the HTTR.

(iii) Technology and components

(iv) Participation in the development of materials optimised for the objectives of the VHTR, beyond the technologies selected around 2006 for the 2020 industrial project. This includes the following steps:
- Optimisation of 9 % chromium steels for the vessel (2008);

— Validation of candidate materials (2008) beyond the super alloys available for the 2020 project, for the intermediate heat exchanger, the primary system and the internal structures of the reactor pressure vessel: Haynes 230 (< 950 °C), ODS (PM2000) and ceramics (SiC) (> 1000 °C);
— Characterization out of pile (2004-2006) and under irradiation in Osiris (2005-2007) of the best selected nuances of graphite;
— Selection (2008) of C/C materials for internal structures in the hot zones of the reactor (control rod sheath);
— Complements in the ASME code for the codification of mechanical design rules of structures at very high temperature (2010);

(v) Participation in the development of compact high temperature heat exchangers using the selected materials (2003-2006) and test of mock-ups on the technological loop from 2007 onward;

(vi) Participation in the development of a conversion system using a gas turbine adapted to the conditions of the VHTR for the co-generation of electricity and hydrogen, and for the production of electricity only. Participation in the development of the turbo-compressor by the development fine modelling of gas flow in the machine, and materials for the disc (Udimet 720, Inconel 792 and CM 247) and for the blades of the turbine;

(vii) Realization of tests of key technologies for very high temperature helium systems in representative conditions on helium test benches and test loops:

(viii) Tests of tightness, tribology, thermal barriers, corrosion, purification of helium (2003-2008); development of the necessary instrumentation to characterize studied phenomena;

(ix) Test of components on the technological loop of 1 MW (Helite) planned in 2006;

(x) Test of components (heat exchangers, turbine) and co-generation modes on the international project of loop system in the range of 20 MW (Hello) from 2009.

(xi) In parallel to the development of the system VHTR dedicated to the production of hydrogen and electricity, various studies, likely to induce specific technology developments, assess the potentialities of this system for other applications:
— Desalination of the sea water in co-generation by distillation with multiple effects (studies led within the framework of the action Eurodesal of the 5th European R&D Framework Programme, probably extended to the 6th, and within the framework of bilateral collaborations with Tunisia and Morocco under the aegis of the AIEA);
— Deep-burning of transuranic elements, in particular minors actinides (Am, Cm), difficult to recycle in PWRs, so as to minimize the residual quantity of long-lived waste generated by the fleet of power reactors in France.

5. PARTICIPATION IN THE DEVELOPMENT OF THE GFR SYSTEM WITHIN THE FRAMEWORK OF THE GENERATION IV INTERNATIONAL FORUM

The CEA is a partner of the United States, Japan and other member countries of the Forum Generation IV for the development of the Gas Fast Reactor (GFR) with fast neutrons and integral recycling of the fuel. The R&D plan of the GFR aims to resolve viability issues by 2015 and to confirm its performances by 2020.

Beyond the developments of high temperature materials, components and conversion systems, which are also necessary for the VHTR, the viability and the performances of the GFR is based:

- On the development of new fuels capable of transposing into the fast neutron spectrum the attractive features of the particle fuels (high temperature resistance, confinement of fission products, reasonable thermal conductivity);
- On the development of processes for the treatment of the spent fuel and re-fabrication for recycling offering sufficient potential to be economic, compact and optionally set-up on the production site;
- On the development of a set of redundant and diversified safety systems capable of managing all the accidents conditions, and in particular the accident of depressurisation.
- The orientations retained for the fuel and the fuel cycle processes were defined in 2001; those relative to the safety systems result from studies performed in 2003. These options are already widely shared with the US National Laboratories within the framework of a bilateral I-NERI action on the system GFR. The participation of the CEA in a program of development more opened to international collaboration from 2004 onward is especially focused on the following developments:

(a) Core, fuel and cycle
  - Selection in 2004 of a reference technology for the fuel and the core materials to be first validated through irradiation tests in experimental reactors (Osiris and Phénix (2005-2007)), and then tested after 2013 within an experimental reactor representative the GFR operating conditions (Experimental Reactor for Technology Developments (REDT));
  - Development, on the basis of experiments in the Atalante hot Laboratory and in the Laboratory for advanced fuel development (LEFCA), of processes for spent fuel treatments and fuel re-manufacturing for recycling, that are sufficiently simple, compact to be economical and likely to be set-up on the nuclear site: pre-selection of processes in 2004, demonstration of scientific feasibility (2008), development of the processes and the associated technologies (2008-2020), demonstration of treatment/re-fabrication of 1 kg of GFR fuel (2012).
  - Report in 2006 on feasibility and safety options for the Experimental Reactor for Technology Development (REDT (30 MWth)) planned to start in 2013;

(b) Design and safety
  - Design studies to select the design options and operating parameters of a reference GFR system (2003);
  - Participation in GFR system studies to resolve technological showstoppers (2015), to confirm the performances of the system (2020) and to prepare the decision for an international demonstrator of GFR;
  - Development and validation of the computational tools for the design of the GFR, beyond the needs identified for the VHTR; critical experiments in Masurca to validate the calculation of the neutronic characteristics of the GFR core (2004-

2005), then that from the configurations of the REDT core to evolution between the first core and the GFR demonstration core (2008-2010);
- Study of GFR's demonstrator international project (reactor and fuel cycle plant) on the horizon 2022 according to the following schedule: selection of options (2008-2010), pre-conceptual design studies (2011-2012), conceptual studies (2013-2014), then final design studies (2015-2017);

(c) Technology and components
- Development of materials and specific components for the GFR system (internal structures of the reactor under fast flux, in and out vessel heat exchangers for the systems of decay heat removal);
- Realization of tests in the conditions of the GFR on helium benches and technological loops for the helium systems at very high temperature;
- Test of components on the technological loop in the range of 1 MW (Helite) (2006)
- Test of components (heat exchangers, turbine) and of operation in nominal and accidental modes on the international project of loop system of 20 MW (Hello) (2009)
- Validation of the safety principles of the GFR on the experimental reactor REDT (2013).

6. SUMMARY OF THE LARGE EXPERIMENTAL FACILITIES PLANNED FOR THE "GAS TECHNOLOGY PATH"

The development of the considered set of gas cooled systems (Modular HTR for 2020, VHTR for the co-generation of hydrogen, and GFR with fast neutrons and integral recycling for enhanced sustainability) requires the following large experimental facilities to be developed in the international frame of the Forum Generation IV:

- The technological helium loop Helite in the range of 1MW planned in 2006 for testing components;
- The helium system Loop (Hello, 20 MW) planned for 2009 to test large components, to simulate normal and abnormal operating transients of gas cooled systems with direct conversion, and to validate code predictions of these operating sequences;
- The Experimental Reactor for Technology Development REDT (30 MWth) intended to validate after 2013 GFR fuels and core materials in representative conditions, and to contribute as much as needed in the international context to validate useful technologies for other gas cooled systems (first selection of REDT options in 2003, conceptual design in 2006, final design in 2009);
- Demonstration in an integrated facility, of the operations on the GFR fuel cycle, in terms of performances and production of waste (2012);
- GFR demonstration reactor around 2022 to conduct representative tests of the reactor, then representative tests of the fuel cycle after 2025:
  - (a) For the reactor: selection of options (2008-2010), pre-conceptual studies (2011-2012), conceptual studies (2013-2014), final design (2015-2017);
  - (b) For the experimental pilot for the cycle: selection of a process of spent fuel treatment (2005), report on the scientific feasibility of the reprocessing/re-fabrication processes (2016), documentation of process (2018), starting of the installation (2025).

7. PARTICIPATION IN THE DEVELOPMENT OF THE OTHER SYSTEMS WITHIN THE FRAMEWORK OF THE GENERATION IV INTERNATIONAL FORUM

The CEA participates in the development of the Sodium Fast Reactor system (SFR) with the objective to share the expertise acquired in France on Phenix and Superphenix, to participate in further investigations on key issues for sodium cooled reactors (in service inspection and maintenance, sodium risks, management of severe accidents), as well as to contribute to the development of key technologies for the Generation IV version of this type of system: minor actinides bearing fuels, supercritical CO_2 gas turbine.

In parallel, the evaluation of the potentialities of reactors cooled with supercritical water and with molten salts, which developed within the framework of European 5th R&D Framework Program, will be extended in 2004 through the participation in the evaluation of Generation IV Super Critical Water cooled Reactor (SCWR) and Molten Salt Reactor (MSR). The CEA will be involved particularly in the system and safety studies of reactors with supercritical water, as well as in system and safety studies, scenarios studies, and R&D on salt treatment processes for molten salt reactors.

8. INTERNATIONAL COLLABORATION AS MAIN PATH FORWARD TO DEVELOP THE GENERATION IV SYSTEMS

As a consequence of the Technology Roadmap in 2000 that led it to focus future nuclear energy systems studies on a consistent set of gas cooled systems, the CEA included R&D items relevant to this “Gas technology Path” in renewed versions of the collaboration agreements with Japan, United-States and Russia. The cooperation in Europe in this field developed in the 5th R&D Framework Program at the initiative of the HTR Technology Network.

The R&D work that has been launched since 2001 as a national program on gas cooled nuclear systems is appealed to further develop within the Generation IV International Forum, as the potential of the considered set of gas cooled systems to meet international market needs in the medium term, to enable efficient hydrogen production in the longer term (VHTR), and to ultimately afford sustainable nuclear production (GFR), was confirmed by the technology roadmap conducted by the Forum.

Besides, CEA will maintain innovative research on the mature technology of sodium cooled reactors through an active participation in the development of the Generation IV Sodium Fast Reactor (SFR). With a lower priority, the CEA will proceed with the assessment of feasibility and performance of more prospective nuclear systems while participating to system studies on the Generation IV Supercritical Water cooled Reactor (SCWR) and Molten Salt Reactor (MSR).

Future work on New and Innovative nuclear Concepts in Europe (6th R&D Framework Programme) is currently being structured so as to permit easy connections with equivalent activities in the Generation IV Forum to maximise the benefit of a broad international collaboration and therefore to secure prospects of successful development of the innovative technologies recognized as most promising for nuclear power to meet 21st century energy needs.

NUCLEAR OPTIONS FOR INDUSTRIALIZED AND DEVELOPING COUNTRIES RESULTS OF INOVACT PROJECT

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Abstract. The main objective of the project is to detect nuclear options which could present an industrial interest for the replacement of the EDF fleet at the two horizons of 2020 and by 2070. In order to provide the best possible answer, the project has to address other criteria than the usual technical-economical one and consider every possible option. In particular, the analysis has to integrate also all the aspects : environmental, political, sociological, and market oriented. A forecast study defined scenarios leading to the description of the energetic situation at horizons 2020 and by 2070. Each scenario must make it possible to describe the energy context (evolution of nuclear energy in the production of electricity in particular) and the impact of this context on nuclear system requirements, through the eight criteria characterizing the concepts. These criteria take into account the global energetic issue, especially the sustainable development. An assessment chart performed from these criteria, and each criterion has been split into sub criteria, allowing us to establish for each innovative concept the level of requirement fulfilment. The most suitable concepts with regard to the scenarios will be those which will fulfil the highest number of selected criterion by reaching the highest requirement level.From these selected concepts relevant to the scenarios, an EDF R&D program will be defined and implemented in order to improve the knowledge and to appreciate the degree of maturity of the concepts.

1. INTRODUCTION

Today, the greater part of electricity generation in France is provided by 58 PWR units. For EDF, it is important on the one hand to continue to operate existing power stations efficiently so as to achieve a mean lifetime of at least 40 years (50 years is currently regarded as a realistic mean), and on the other hand, to prepare for the future with the aim of progressive replacement of the facilities. The lifetime parameter will contribute to the competitiveness of the PWR system, which will depend on the price of other forms of energy, including gas in particular, in the years to come. Furthermore, this could offer hope for the emergence of new technologies, based on more efficient fissile and fossil fuels or renewable forms of energy.

Preparation for the future will be based on relatively sophisticated projects, which could be of industrial relevance for 2020 and approximately 2070 horizons. For this purpose, research in the field of new nuclear systems will have to take a number of criteria, apart from the basic technico-economic criterion, into account. These include :

— competition from different types of energy, intensified by the opening of the European electricity market,
— plant safety,
— electricity market structure, and the share targeted by or accessible to EDF,
— energy resources (including oil and gas in particular),
— international commitments as regards the reduction of greenhouse gas emission,
— waste management (back end of the fuel cycle in the case of nuclear energy),
— public opinion,

— governmental decisions,
— international regulations.

Forecast evolution of energy demand, installed nuclear production capacity, the competitiveness of nuclear energy and the limitation of greenhouse gases work in favor of the nuclear system, at least in terms of basic production. Furthermore, the prospects of extended service life for existing power plants will also enhance their competitiveness, and create the need for maintaining expertise in the areas of operation and research.

In addition, in order to ensure that "the nuclear option remains open", the nuclear energy industry of the future is to be competitive and acceptable by public opinion. Current technological progress points the way to potential systems with enhanced performance in terms of thermal efficiency (the maximum figure for PWR units is 35%), burnup fraction and waste management. EDF must be in a position to quantify the relevance of these new options.

The major strategic issue for EDF is to ensure that the nuclear option remains open. EDF must contribute its industrial vision of future systems in its capacity as operator, and thus orient public research conducted in this domain. EDF must also be able to express a critical opinion concerning all potential solutions, and identify the most promising.

Prospective actions aimed at proposing a new nuclear system must consequently be based on the expertise of widely differing individuals who will contribute technical, social, economic, safety, operating, energy-related and environmental viewpoints, so as to avoid any preconceived notions in the approach.

The main objective of the INOVACT project is to identify one or more nuclear systems which should be relevant from an industrial point of view for 2020 and approximately 2070 horizons.

2. IMPLEMENTATION OF THE PROJECT

The INOVACT project is planned for a period of 3 years from 1 January 2001 to 31 December 2003.

The project is structured in a number of principal work-packages as follows:

— *State of the art:* this segment covers collation of all available knowledge primarily relating to fuel, reactors and waste management for the various system. This segment carried out the full period of the project.
— *Forecast study:* The forecast study has been implemented during the year 2001, the aim being to define a set of scenarios which will describe energy demand situations and electrical systems for 2020 and approximately 2070 horizons.
— *Specifications and assessment chart:* Drafting of specifications for the system or systems corresponding to the various scenarios (one set of specifications per scenario), in which nuclear energy will have a significant position, has been performed during the year 2002, following by an assessment chart, integrating specification discriminatory criteria among others.
— *Proposal and assessment of nuclear concepts:* Comparative assessment can only be relevant on the basis of in-depth knowledge of the various systems. A nuclear engineering consultant firm charged with the study so as to collate the maximum amount of data. The work involved definition and comparison of the various known systems using input data from the state of the art work-package. A comparative

assessment table for the various systems established by end 2002 allowed us to select the most suitable concepts.

— *Roadmap:* From the previous selected concepts, an EDF R&D program will be defined for 2003 and will achieve beyond the end of the project in order to improve the knowledge and to appreciate the degree of maturity of the concepts. This program is gathering items and resources needed to develop the new nuclear energy systems.

General articulation of the various work-packages is illustrated in Figure 1.

3. FINDINGS OF THE PROJECT

3.1. Findings of the forecast study

3.1.1. Purpose and scope of the forecast study

The purpose of the forecast study is to construct a set of scenarios which describe the energy contexts of the future (in particular in relation to electrical systems), which are discriminatory in terms of nuclear system[1] specifications. This means that each scenario must make it possible to describe the energy context (evolution of nuclear energy in the production of electricity in particular), and the impact of this context on nuclear system requirements. The scope of the study is world-wide, and the problem has been broken down into two parts, covering industrialized countries and developing countries[2]. Two working horizons have been adopted – 2020 and approximately 2070 – which could correspond to the first and second renewal phases for existing nuclear facilities respectively.

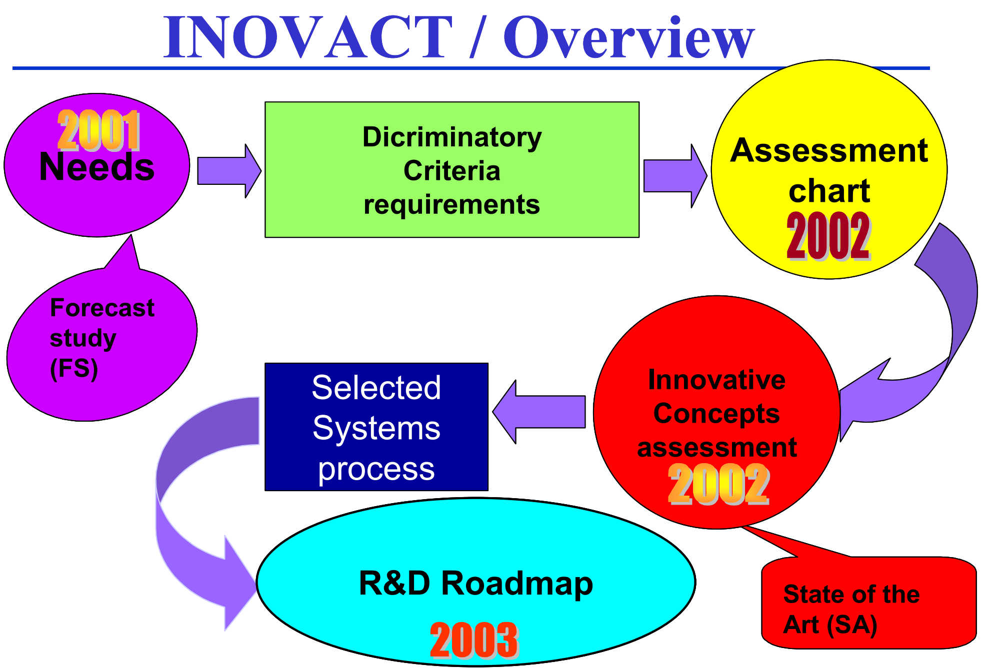

FIG. 1. General articulation of the various work-packages.

[1] The term "system" covers fuel, reactor, waste and the back end of the fuel cycle.

[2] This breakdown was only decided during the study. It became apparent that the characteristics for industrialized countries and less developed countries differed too greatly in terms of energy systems by 2020 horizon for it to be possible to address them together in a single model.

The purpose is to present the widest range of possible future situations with the aid of these three scenarios, one of which is very probable (basic scenarios), another probable but different from the basic scenario (contrasted scenario), and the third very unlikely but with strong implications if it should occur (breaking scenario).

3.1.2. Industrialised countries – 2020 horizon

3.1.2.1. A1: "Limitation of renewal to a number of existing sites" (basic)

The public is extremely fearful of the radiological hazards of the systems, resulting in poor public acceptance of nuclear energy as a major accident has occurred before 2020. In this context the opening of new sites is extremely difficult. The highly voluntarist integration of sustainable development, and the sharp rise in energy consumption in the world generates a strong demand for a reduction in greenhouse gas emission. In addition, prices for fossil fuels rise sharply[3].

As the competitive advantage factor is highly favorable for nuclear energy (no competition from other energies and drastic reduction of greenhouse gases), and as it is also practically impossible to open new sites, the level of nuclear energy production in the industrialized countries remains stable, with renewal of the production facility on existing sites.

3.1.2.2. A2: "Upturn for nuclear energy in all unit sizes" (contrasted)

Nuclear energy production in the industrialized countries is substantially greater in 2020. This is explained by the competitive advantage of nuclear energy compared with other competing production methods, insofar as the price of fossil fuels increases sharply, and the objective of drastic reduction of greenhouse gases is declared at political levels in line with sustainable development aims. These favorable elements are reinforced by excellent acceptance by the general public, which is in favor of the opening of new sites as no major accident has occurred between now and 2020.

3.1.2.3. A3: "Gas systems" (breaking)

Major new discoveries of oil and natural gas resources result in stabilization of fossil fuel prices. Nuclear energy is consequently faced with strong competition. This competition is also intensified as a result of abandonment of greenhouse gas reduction policies. Sustainable development is no longer considered as a major issue[4]. Nuclear energy production is thus reduced, despite good public acceptance of this energy form (no major accident has occurred before 2020). Non-integration of the notion of sustainable development and the resultant low level of aid for renewable energy sources allow massive development of the gas system without any thought of reducing greenhouse gas emission.

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[3] Prices for fossil fuels rise because the discovery rate for new gas and oil resources is still below world consumption. CO2 sequestration techniques are also extremely costly.

[4] This relative disinterest in the notion of sustainable development is explained, in particular, by the fact that no increase in the frequency or severity of natural catastrophes has been observed.

3.1.3. Industrialised countries – 2070 horizon

3.1.3.1. B1: “Plant renewal primarily on renovated existing sites” (basic)

There is a poor public acceptance of nuclear energy and programs for economizing energy have launched. At the same time, the political decisions aim at reducing the greenhouse effects. As a result the nuclear generation is stable. The plant renewal is implemented with large size of reactors asking to sustainable criteria.

There is no contrasted scenario for this case. It is the same as basic scenario.

3.1.3.2. B3: “Nuclear energy is slightly increasing with current constraints” (breaking)

The nuclear energy public acceptance is very strong due to the good nuclear management over the world (no accident, remediation at time). In that scenario there is no political willingness of sustainable development. Thus, the nuclear generation is moderately increasing. In addition, the electricity market is completely open. The new nuclear systems will have to fulfill high competitive constraints.

3.1.4. Developing countries – 2020 horizon

3.1.4.1. D1: "Gas systems and renewable energy sources"

Nuclear production is reduced, or has not commenced in the DCs. The nuclear image is negative due to the fact that a major accident has occurred in the world before 2020. Fossil fuel prices only increase moderately, in particular due to the discovery of new oil and gas reserves. In addition, transport networks in the DCs have not been developed to any great extent[5]. Thus energy-related decisions, which then take full account of the problems of sustainable development and reduction of greenhouse gases, are directed principally at the replacement of coal- and oil-fired systems by gas systems, associated with development of renewable energy forms. Given all the constraints to which it is subject, nuclear energy is not regarded as a satisfactory solution.

3.1.4.2. D2: "Emergence of small reactors" (contrasted)

Despite a favorable competitive advantage (sharp increase in fossil fuel prices, and greenhouse gas reduction policies) combined with good public acceptance (no major accident has occurred between now and 2020), the level of nuclear energy production in the DCs is stable. This is largely explained by the unstable world political situation, making the opening of new sites extremely difficult, and by the continued low level of development of transport networks.

In this context, only decentralized production facilities can be considered. To meet a sharp increase in energy demand in the DCs, the "energy mix" solution appears appropriate, with the possible emergence of small reactors with low fuel consumption and proliferation levels.

[5] The weak financial resources of the DCs are assigned principally to decentralized production rather than the development of networks.

3.1.4.3. D3: "Large reactors" (breaking)

Nuclear energy is in a strong position to cater for the vast increase in energy demand in the DCs. Public acceptance is good (no major accident has occurred between now and 2020), the competitive position is favorable (fossil fuel prices on the increase, and clearly declared greenhouse gas reduction policies), transport networks have undergone considerable development and the international political situation is tending to stabilize.

In this case there is a trend towards a majority share for nuclear energy in the "energy mix", with large reactors (developed network) used to meet the sharp increase in energy demand, as also environmental protection considerations whether in terms of waste management or fuel economy (closed cycle).

3.1.5. Developing countries – 2070 horizon

3.1.5.1. E1: "Plant renewal limited to renovated existing sites" (basic)

The nuclear generation is stable. That situation is due to two antithesis effects. The nuclear competitive advantage (high other fuel prices, greenhouse effect reduction) is balanced by the extreme difficulty to open new sites (an accident has occurred before 2070, and the public acceptance is very low). That scenario is quasi equivalent of A1 one.

3.1.5.2. E2: "Nuclear energy increasing" (contrasted)

Nuclear energy production is dramatically increasing. This situation is due to the competitive advantage of nuclear energy compared with other competing production methods, insofar as the price of fossil fuels increases sharply, and the objective of drastic reduction of greenhouse gases is consistent with political decisions in favor of sustainable development. These favorable elements are reinforced by excellent public acceptance. As a result, the opening of new sites is encouraged.

3.1.5.3. E3: "Coal systems" (breaking)

There is no political willingness of sustainable development and the fossil fuel prices increase sharply. The best solution to answer the consumption demand is the coal plants.

As a conclusion, among the eleven scenarios which have been drawn up for industrialized and developing countries (ICs and DCs) by 2020 and approximately 2070, eight scenarios correspond to nuclear energy.

3.2. Findings of the assessment chart

A list of discriminatory criteria (and their expected requirements) has been prepared for drawing up the specifications for future nuclear systems (Table I below). This list thus constitutes the start point of the chart.

Table I. List of the 8 discriminatory criteria

Aim of a competitive price per kWh	highly competitive
Safety requirement for the operator	strengthening
Possibility of proliferation	with difficulty or no
Rational fuel utilisation	search for savings
Operability	high
Technico-organizational and prescriptive structure	simple
Constraints relating to waste minimization	high
Requirement for a passive system	yes

Each criterion has been split into several other criteria in order to evaluate the innovative concepts. There are 62 criteria required for the appraisal. Different weights are attributed to the sub criteria to calculate the global note associated to each discriminatory criterion.

3.3. Findings of the innovative concept assessment

3.3.1. Methodology

The innovative reactor concepts assessment aims at performing a comparative classification of different nuclear systems (reactor and fuel cycle) in connection with the eight scenarios of the forecast study which chose nuclear energy as a relevant solution to solve the energy consumption in the future.

The list of the fourteen evaluated concepts get together all the families of concepts put into the different coolant categories ; the light water systems (PWR, BWR, CANDU NG, Supercritical), liquid metal systems (sodium or lead), gas and molten salt systems. Otherwise, these concepts are also classified into three groups characterising their degree of industrial maturity as measured today. Industrial maturity means the commissioning of a demonstrator plant followed by several operation years and a significant feedback experience. This time span represents ten years or more for the most innovative concepts. The first group (G1) is considering the concepts available by (2010-2020), the second (G2) by (2020-2030), the last (G3) is gathering the most innovative concepts namely those with the highest uncertainty and as a result the highest range (2030-2070). The concepts are as following :

— G1 = {AP 1000, EPR UOX, ESBWR, SWR 1000, ACR 700}
— G2 = {BN 800, BREST 1200, EFR, EPR APA, GT-MHR, IRIS}
— G3 = {AMSTER, GCFR, SCWR}

A radar graph is associated with each scenario which indicates the level of expected requirements on the eight specification criteria. These criteria are the economic competitiveness, the safety, the proliferation resistance, the fuel utilisation, the operation of the facility, the structure to manage a severe accident, the waste minimization and the passive systems. The size (electrical power) is an exclusive criterion which is not represented on the graph.

These criteria have not the same importance for assessing the systems. In order to establish a global note for each concept a weight has been given to each criterion. These weights are the same for the developed countries by 2020 and 2070 and for the developing countries by 2070 that should be considered as developed countries by 2020. In this case, the most important criteria are the competitiveness, the safety, the fuel and the wastes. For the developing countries by 2020 the weights are strong on the competitiveness, the proliferation resistance, the safety, the passive systems, the fuel and the operation of the facility. The concepts are evaluated between them in comparison of the degree of reached requirement on each criterion. The notes given to the criteria characterize the concepts and the weights attributed to the notes characterize the scenarios. Therefore, with the same notes, a concept does not take the same place on each scenario. It depends on the degree of requirement asked for each specification criterion.

3.3.2. Industrialised or developed countries

Thus for the developed or industrialized countries as France, the finding of the comparative evaluations by 2020 are the following. Concerning the basic scenario characterized by a stable evolution of the nuclear generation and a replacement of the old plant on the existing sites, the systems will have to be in accordance with high requirements on all the criteria (illustrated with Fig. 2) and a size of 1000 MWe and over. However the requirement on the competitiveness is moderated because the prices of the other fuels are very high and there is a political willingness to reduce the greenhouse effect, that is in favour of the nuclear option. The relevant concepts belonging to the group one G1 are the evolutionary or passive PWR and BWR (EPR UOX, AP 1000, SWR 1000, ESBWR). Those of the group two G2 are primarily LMFR (Liquid Metal Fast Reactor), sodium or lead (EFR, BN 800, BREST 1200), but also the HTR (GT-MHR) with the hypothesis that it can reach the expected power by adding several modular concepts with no more required ground area than a 1000 MWe PWR. The group three G3 concerns only the MSR (AMSTER). The needs of this scenario are fulfilled with the G1 and G2 systems.

The second scenario, the opposite scenario, is characterized by an increased nuclear generation with all the range of possible sizes. It is the same scenario as the GEN IV scenario (sustainability) in which the requirements are very strong on only two criteria namely the fuel and the wastes and moderated on the others (Fig. 3). There is no G1 concepts answering well these high requirements. In the G2 group there are exclusively the LMFR (BREST 1200, BN 800, EFR) and the G3 group gathers the GCFR and the MSR (AMSTER). But to this horizon (2020), the R&D efforts should be very significant to reduce the G3 maturity time.

By the 2070 horizon the basic scenario is characterized by the replacement of old plants primarily on the renovated sites. It is close to the basic scenario at 2020 and the suitable systems are the same (Fig. 4). The needs of this scenario are fulfilled with the G2 (BREST 1200, EFR, BN 800, GT-MHR) and G3 (AMSTER) systems. The last scenario concerns a slight nuclear growth with the same usual constraints as today, but with reinforced requirements on the fuel and the competitiveness criteria (Fig. 5). The competitiveness is due to the total opening electricity market. In addition there is no sustainability willingness. The suitable concepts will be with a size of 1000 MWe and over in order to be more competitive. The needs of this scenario are fulfilled with the G2 (GT-MHR) and G3 (SCWR) systems.

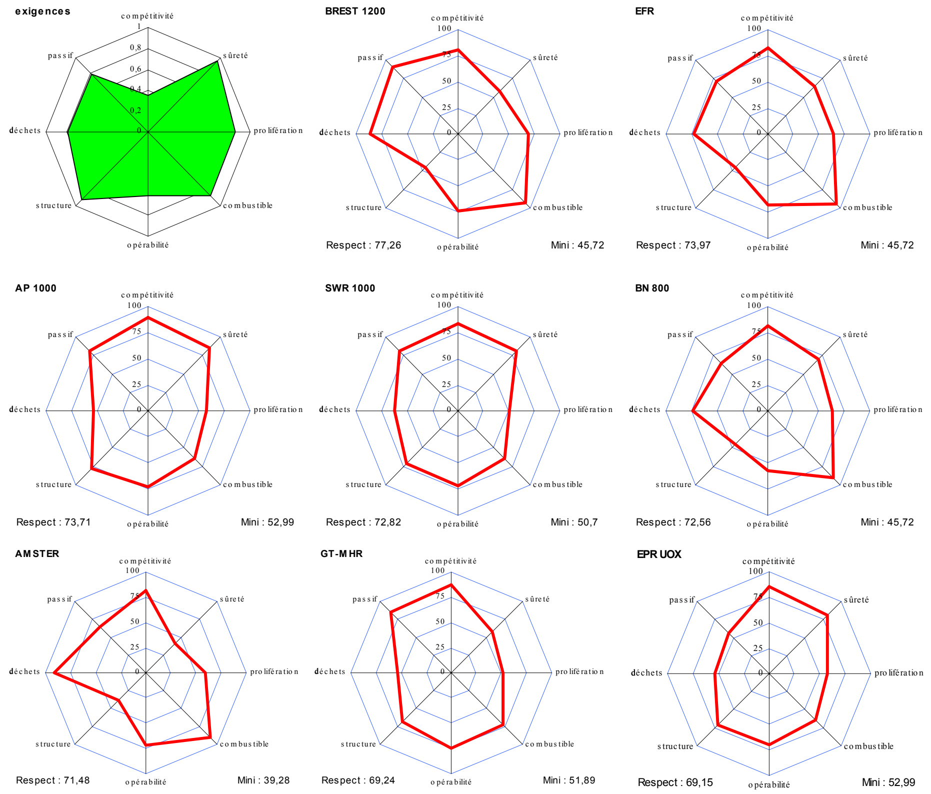

FIG. 2. Developed countries – 2020, basic scenario A1. High requirement level on each criterion.

FIG. 3. Developed countries – 2020, contrasted scenario A2, relevant with GEN IV scenario (sustainability).

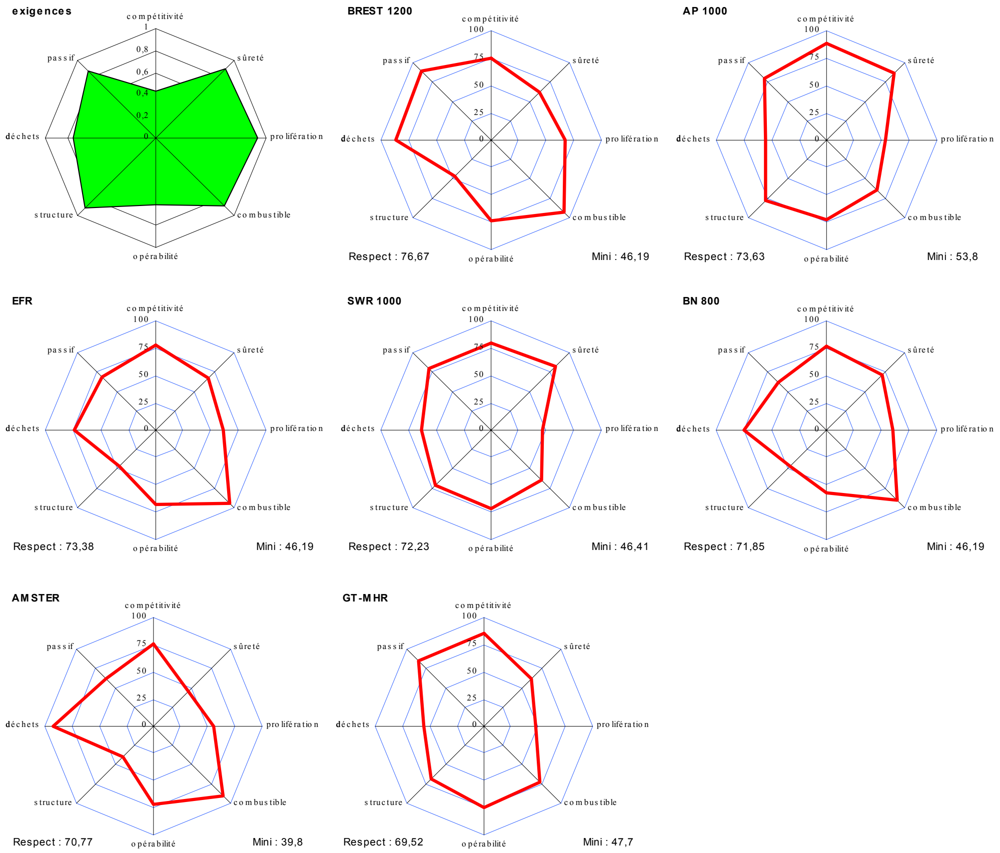

FIG. 4. Developed countries 2070 – basic scenario B1, close by A1.

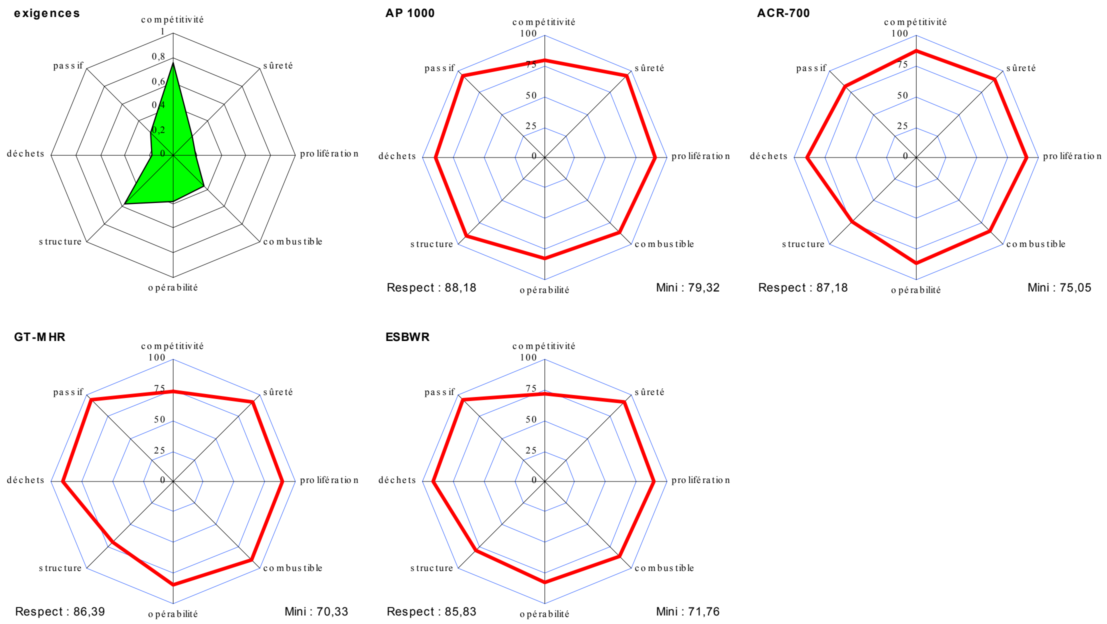

FIG. 5. Developed countries 2070 – breaking scenario B3.

3.3.3. Developing countries

For the developing countries, the finding of the comparative evaluations by 2020 are the following. Concerning the first scenario, contrasted scenario, it is characterized by a stable evolution of the nuclear generation and the emergence of small size concepts, the systems will have to be in accordance with high requirements on two criteria, no proliferation and rational use of fuel (illustrated with Fig. 6). The relevant concepts belong only to the group two G2 as IRIS concept.

The second scenario, the breaking scenario, is characterized by an increased nuclear generation with large sizes of reactors. It is the same scenario as the GEN IV scenario (sustainability) in which the requirements are very strong on only two criteria namely the fuel and the wastes and moderated on the others (Fig.7). There is no G1 concepts answering well these high requirements. In the G2 group there are exclusively the LMFR (BREST 1200, BN 800, EFR) and the G3 group gathers the GCFR and the MSR (AMSTER). But to this horizon (2020), the R&D efforts should be very significant to reduce the G3 maturity time.

By the 2070 horizon the basic scenario is characterized by the replacement of old plants primarily on the renovated sites. It is close to the basic scenario at 2020 and the suitable systems are the same (Fig. 8). The needs of this scenario are fulfilled with the G2 (BREST 1200, EFR, BN 800, GT-MHR) and G3 (AMSTER) systems. The last scenario concerns a slight nuclear growth with reinforced requirements on the fuel and the waste criteria (Fig. 9). The suitable concepts will be with all kind of size. The needs of this scenario are fulfilled with the G2 (BREST 1200, BN 800, EFR) and G3 (AMSTER, GCFR) systems. It is the same scenario as the GEN IV scenario (sustainability).

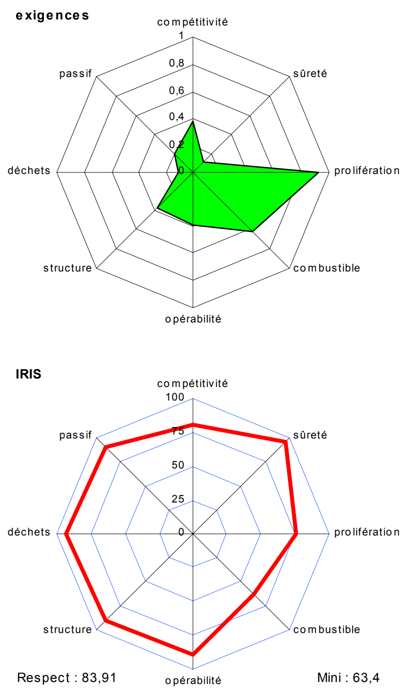

FIG. 6. Developing countries 2020 – contrasted scenario D2, small size reactors.

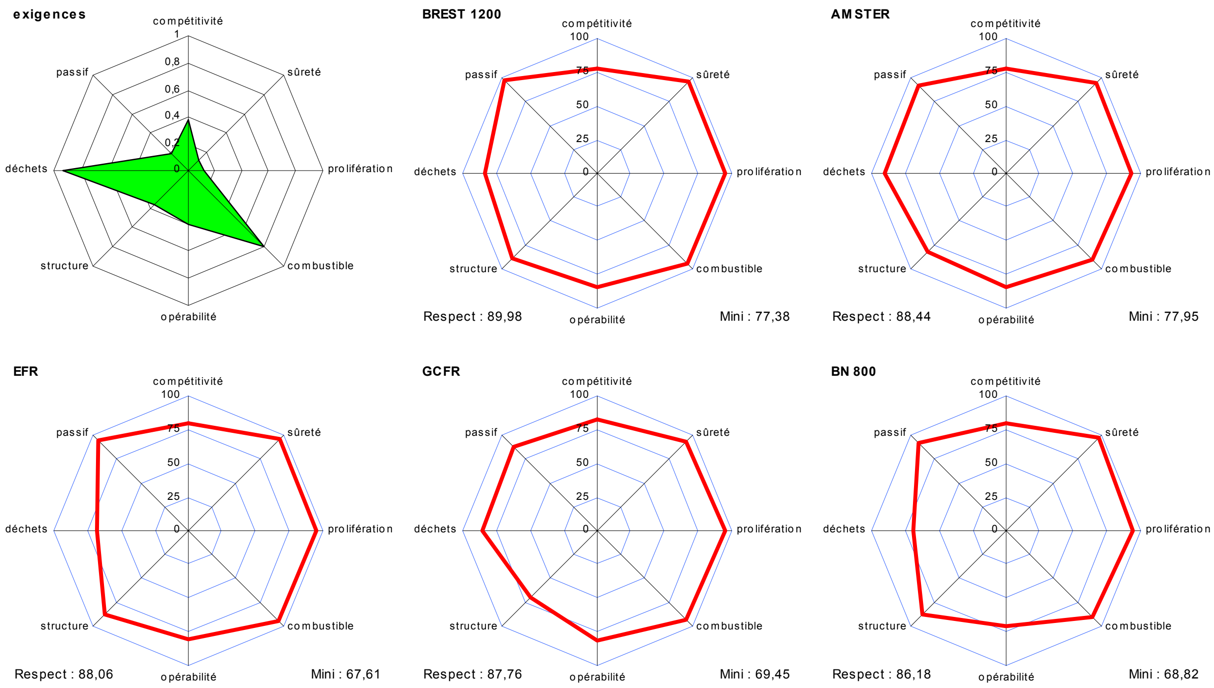

FIG. 7. Developing countries 2020 – breaking scenario D3, relevant with GEN IV scenario (sustainability).

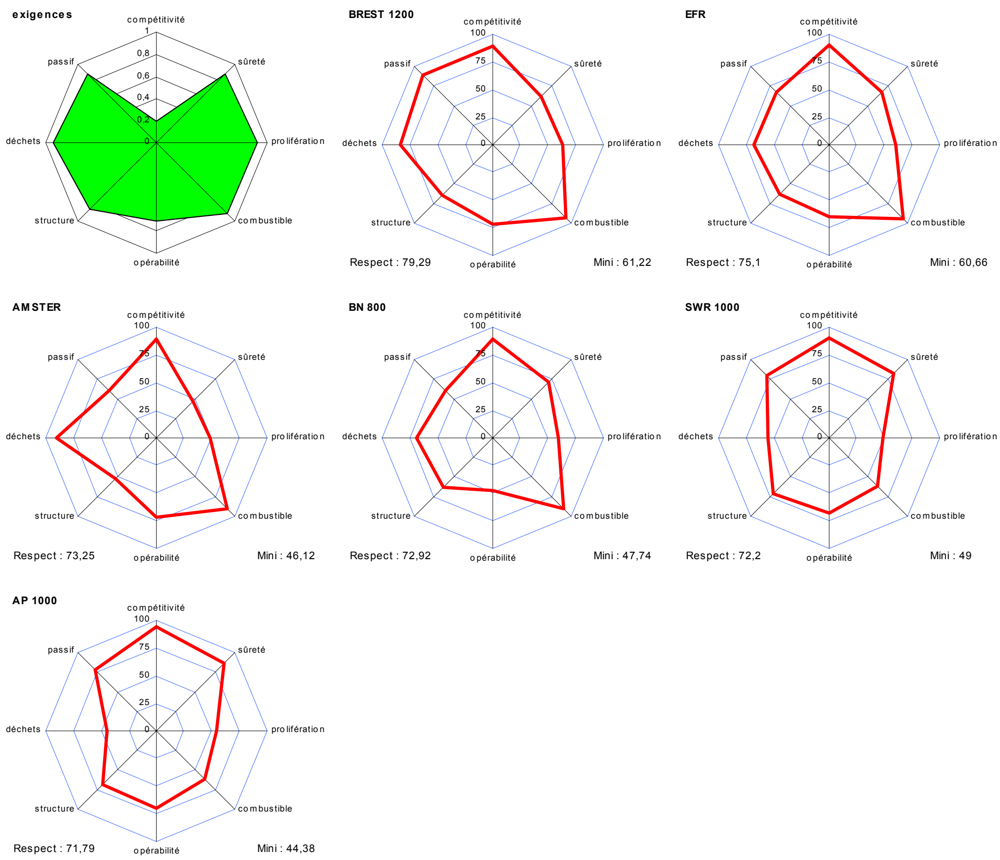

FIG. 8. Developing countries 2070 – basic scenario E1, close by A1.

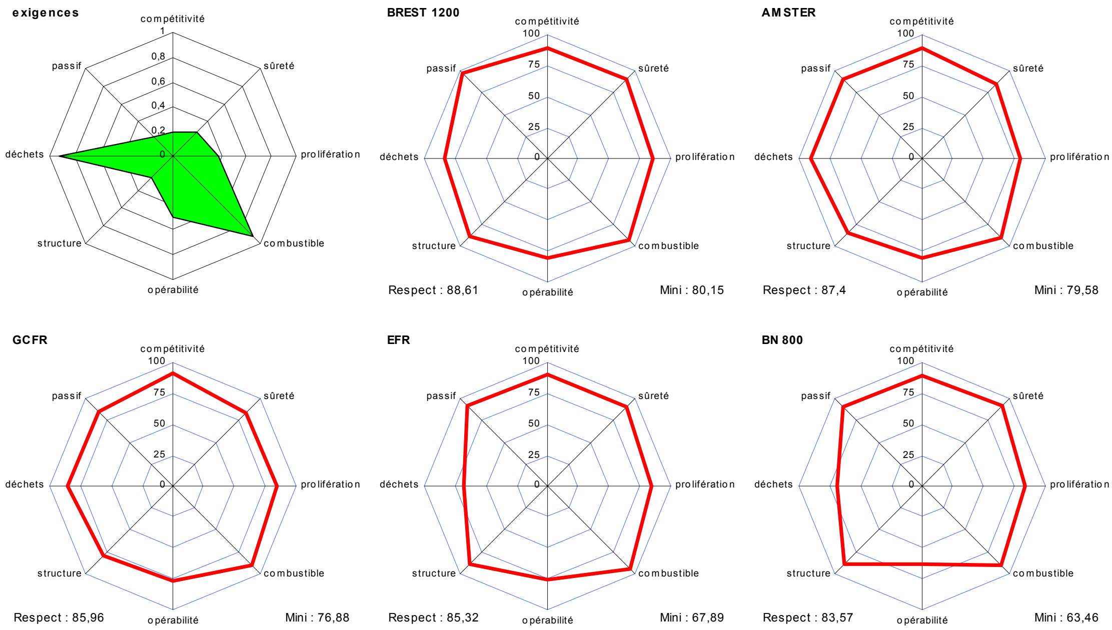

FIG. 9. Developing countries 2070 – contrasted scenario E2, relevant with GEN IV scenario (sustainability).

4. CONCLUSION

The findings of the evaluations done in INOVACT project are consistent with those of GENERATION IV. The scenarios characterized by the sustainability (very high requirements on fuel and wastes, moderated on the others) select nearly the same systems. The LMFR (Liquid Metal Fast reactor) is always at the top of the list followed by the MSR and then the GCFR. Nevertheless, two concepts do not stand out, the SCWR and the GT-MHR because they are far from the other concepts in comparison with the requirements on the most important criteria (delta of 30% on the waste criteria and from 10% to 20% on the fuel criteria). The VHTR dedicated to hydrogen generation has not been selected in the list of assessed concepts in INOVACT. So it does not appear in the findings.

The last milestone of the project consists in proposing a R&D program on the selected concepts. This program will include the GCFR that is studied by the French Atomic Energy Laboratory (CEA).

The goal of the last work package, "Deployment (DE)", to be able to give a clever judgment on the industrial maturity of the systems from the G2 and G3 groups by unlocking the identified technical gaps. It is scheduled during the first 2003 semester. The deliverable will be a technical note which will describe the items to implement beyond the INOVACT project. The work package "State of the Art" will continue after the end of the project.

5. REFERENCES

[1] Dynamique des systèmes et méthodes prospectives, Travaux et Recherches de Prospective, n°2 – Mars 1996
[2] PEARL, J., Graphical models for probabilistic and causal reasoning, The Computer Science and Engineering Handbook,– 1997
[3] Energy Report 2010 – 2020 – Commissariat Général du Plan – 1998
[4] Global Energy Perspectives – IIASA – WEC, Cambridge University Press – 1998
[5] MURPHY, K., A Variation Approximation for Bayesian Networks with Discrete and Continuous Latent Variables,– UAI 1999
[6] DESSUS, B., PELLAT, R., Economic and forecast study of the nuclear systems – French documentation, JM. Charpin,– September 2000
[7] GODET, M., Creating Futures : Scenario Planning as a Strategic Management Tool, Economica - Brookings, n°2,– 2001
[8] BOOTZ, JP., Prospective et apprentissage organisationnel. Travaux et Recherches de Prospective, n°13,– Janvier 2001
[9] CHATELAIN, C., BOOTZ, JP., LORIETTE, P., Prospective « Réacteurs du futur », projet INOVACT, note HE-21/02/01 et 02,– avril 2002
[10] LORIETTE, P., Elaboration du cahier des charges et de la grille d’évaluation des filières du futur, note HI-28/2002/010/A, – juillet 2002
[11] LORIETTE, P., Evaluation des concepts innovants, Sélection des filières émergentes, note HI-28/03/013/A,– avril 2003

INNOVATIVE CONCEPT OF REDUCED-MODERATION WATER REACTOR (RMWR) FOR EFFECTIVE FUEL UTILIZATION THROUGH RECYCLING

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Abstract. An innovative water-cooled reactor concept named Reduced-Moderation Water Reactor (RMWR) is under development by JAERI in cooperation with utilities and vendors, aiming at effective fuel utilization through plutonium multiple recycling based on the well-experienced water-cooled reactor technology. The reactor is able to achieve a high conversion ratio more than 1.0 with MOX fuel from the sustainable Pu recycling point of view. Such a high conversion ratio can be attained by reducing the moderation of neutrons, *i.e.* reducing the water fraction in the core. Another important core design target is to achieve the negative void reactivity coefficient as in LWRs, especially from the safety point of view. Although the negative void reactivity coefficient and the high conversion ratio are in the trade-off relation in this type of reactor design and this gives difficulty in the design, we have succeeded in obtaining core design concepts satisfying them simultaneously. Especially from the Pu recycling point of view, the core performances during the Pu multiple recycling have also been investigated for the advanced fuel reprocessing schemes with low decontamination factors than the current PUREX process, and are shown to achieve the conversion ratio more than 1.0 and the negative void reactivity coefficient Other detailed investigations have been also performed on the core design, in conjunction with the other related studies such as on the thermal hydraulics in the tight-lattice core including the experimental activities. The experimental results on the critical hea flux have indicated promising results for the core cooling in the tight-lattice core.

1. INTRODUCTION

For the sustainable energy supply with the nuclear power, effective fuel utilization through fuel recycling is a key point. From this point of view, an innovative water-cooled reactor concept named Reduced-Moderation Water Reactor (RMWR) is under development by JAERI in cooperation with some Japanese utilities and vendors, aiming at effective fuel utilization through Pu multiple recycling based on the well-experienced water-cooled reactor technology [1]. The reactor is able to achieve a high conversion ratio more than 1.0 with MOX fuel from the sustainable Pu recycling point of view. Such a high conversion ratio can be attained by reducing the moderation of neutrons, *i.e.* reducing the water fraction in the core. The reduced neutron moderation with the water results in a similar neutron spectrum to that in a sodium-cooled fast breeder reactor (FBR) even in a water-cooled reactor core as shown in Figure 1.

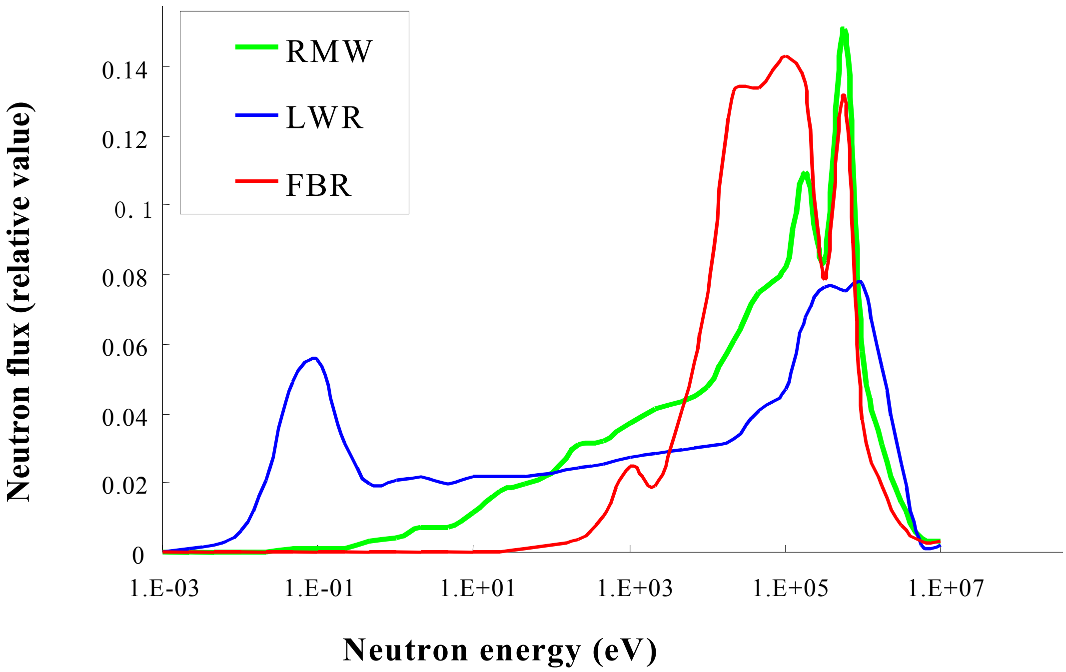

FIG. 1. Comparison of nutron spectra among RMWR, LWR and FBR.

The high conversion ratio is favorable to maintain Pu quality, *i.e.* fissile Pu percentage and makes it possible to recycle Pu many times, *i.e.* Pu multiple recycling. This is a preferable method for deployment of the reprocessed Pu from UO_2 spent fuel, and also contributes to reduction of the spent fuel accumulation. Therefore, high conversion ratio is one of the main targets for designing RMWR core. Another important design target for the RMWR core is to achieve the negative void reactivity coefficient. This is one of the important characteristics of the currently operated LWRs, especially from the safety point of view. However, the negative void reactivity coefficient and the high conversion ratio are widely known to be in the trade-off relation in the reactor design, and this gives a difficulty to be overcome in RMWR designing.

Up to the present, we have proposed several types of design concepts [2, 3] satisfying both the main design targets mentioned above under both the BWR-type concept and the PWR-type one. The common design characteristics are the tight-lattice fuel rod configuration and the short core. The former is to attain the high conversion ratio and the latter is for the negative void reactivity coefficient. Additionally, the axial, *i.e.* upper, lower or internal, or the radial blankets made of the depleted UO_2 (DU) are also introduced by necessity for both purposes mentioned above.

For the RMWR utilization, multiple recycling of Pu is the essential basis under the MOX spent fuel reprocessing cycle. On this point, the fuel cycle with some advanced and economical reprocessing schemes should be taken into account. The typical reprocessing schemes proposed for FBRs might be adopted also for RMWR. The advanced reprocessing schemes currently proposed for the FBR fuel cycle are, however, in general with the lower decontamination factors (DFs) than the current PUREX process and intentionally allow some amount of fission products (FPs) and minor actinides (MAs) to remain in the MOX fuel. Since they are expected to give negative effects on the core performances and the conversion ratio of RMWR is generally just more than 1.0, it is necessary to investigate the core performances of RMWR under the multiple recycling situation with such the advanced fuel reprocessing schemes in relatively low DFs. Main results on this issue are presented in the following as well as the results from recycling under another economical reprocessing scheme even with relatively high DFs proposed by JAERI.

Furthermore, other detailed investigations on the RMWR core, such as control rod operation planning and improvement of core performances have been also performed in the core design. The other related necesarry studies such as on the thermal hydraulics in the tight-lattice core are essential and have been performed including the experimental activities.

2. DESIGN CONCEPT OF RMWR CORE

2.1. General aspects in RMWR core design

The main design goals for the RMWR core are the high conversion ratio more than 1.0 and the negative void reactivity coefficient as already mentioned above. In order to achieve the high conversion ratio, the volume of the moderator, *i.e.* water, should be reduced. For this purpose, the tight-lattice fuel rod arrangement is commonly adopted. The triangular lattice with a narrow gap between the fuel rods and/or the rods with a large diameter is typical for it. Especially for the BWR-type reactor design, increase in the core void fraction is another realistic technique to be used. To satisfy the requirement of the negative void reactivity coefficient, neutron leakage should be increased when the void is generated or increased in the core. The short core design is common technique for it. The blanket regions could also be adequately used to increase neutron absorption. Some streaming mechanism might be also used to promote neutron leakage effect.

The above two design goals are attained by appropriately combining the basic techniques described above, and by keeping a balance among them. In our conceptual design study, some different core designs have been investigated based on the different basic ideas described above. That is, there are some design possibility in adopting and combining the above basic ideas. Up to the present, we have succeeded in proposing several types of basic design concepts [2, 3], satisfying both the main design targets for the conversion ratio and the void reactivity coefficient, under both the BWR-type concept and the PWR-type one. It has, however, been recognized that BWR-type concepts have some advantages in design to achieve higher core performances than PWR-type concepts, because the water fraction can be reduced in the former by utilizing void in the water without increasing tightness of the core. Therefore, the representative RMWR core design is presently set based on the BWR-type concept as described more in detail in the following with the same system configuration as the current ABWR design.

2.2. Representative design for RMWR core

The representative 1,356 MWe class large-scale BWR-type core design is schematically shown in Figure 2. In order to achieve negative void reactivity coefficients, the core is designed to be short and flat, which is suitable to increase neutron leakage from the core. Axial blankets with DU are also introduced to increase the conversion ratio and to reduce the void reactivity coefficients. In the present design, two MOX core layers are shortened to about 200mm high. An internal blanket region of 295mm is positioned between the two MOX layers. With the upper and lower blankets, the total core region has five-layer structure with a total height of 1.105m. As shown in the figure, the pitch of the assembly is 228mm. The diameter of the fuel rods and the gap between rods are 13.7mm and 1.3mm, respectively.

The major dimensions and characteristics of the core are summarized in Table I. The average fissile Pu content in the MOX parts is 18%, and the average fissile Pu content in the core region including MOX and internal blanket parts is 10.4%. The fuel assembly consists of five kinds of fuel rods with different Pu contents to reduce the radial power peaking factor. Y-shaped control rods with the follower structure are introduced for every three fuel assemblies. The core average void fraction is increased to 70%. Therefore, the effective volume ratio of the water to the fuel (*Vm/Vf*) is reduced extremely to about 0.17 in the present design, attaining a high conversion ratio. The resultant conversion ratio of fissile Pu is 1.05.

And the void reactivity coefficient is evaluated to be negative value of -0.5×10^{-4} dk/k/%void. The core can be cooled by natural circulation because the core pressure drop is as low as 0.04 MPa due to the low flow velocity and short core height. Therefore, the reactor internal pumps used in the ABWR can be eliminated in the present design. The number of fuel assemblies is 900 and each assembly has 217 fuel rods. The core outer diameter is 7.6m. The core average discharge burn-up including the internal blanket is 60GWd/t. The operation cycle length is 24 months.

In the neutronics calculations, the group constants of the assembly are prepared by the Monte Carlo method coupled with the burn-up calculation in 190 groups. The core calculation was performed by the three-dimensional void-power iteration method, treating each assembly separately. Another design for a 300 MWe class small-scale core has been also accomplished under the similar core concept [4].

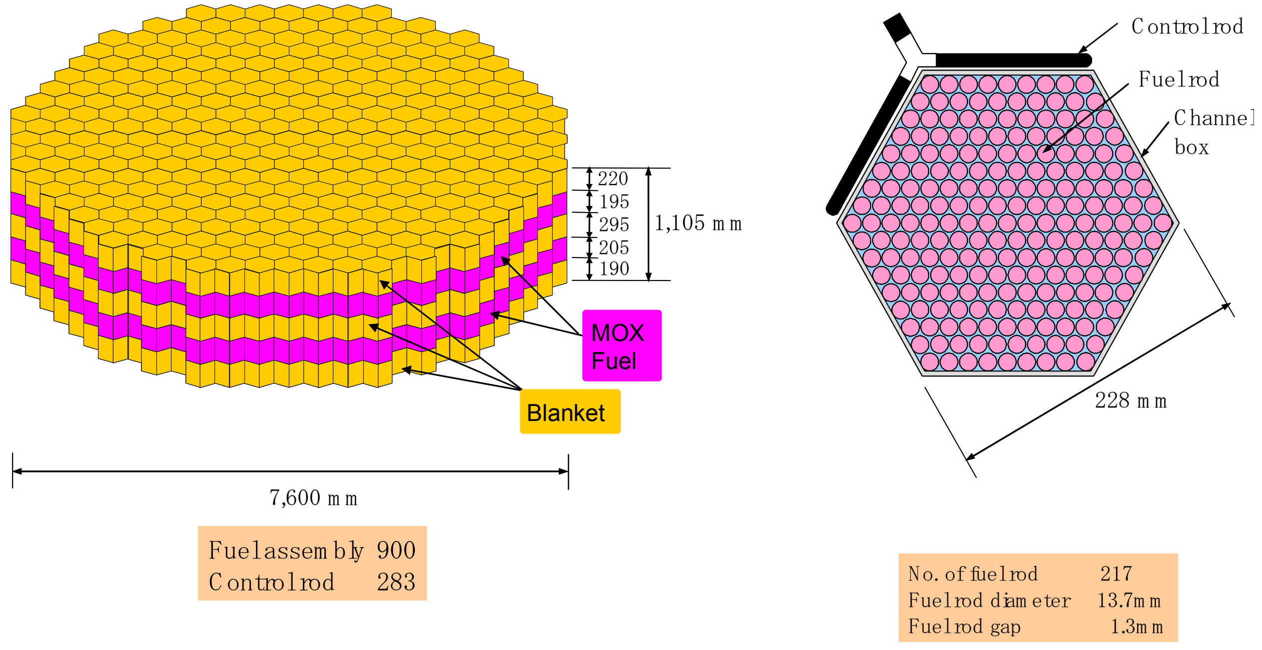

FIG. 2. Schematic of representative core (1,356MWe) and fuel assembly cross section.

Table I. Core major dimensions and characteristics

Item	Unit	Design value
Electric power output	MWe	1,356
Core circumscribed radius	m	3.8
Core average burn-up	GWd/t	60
Core effective height	m	0.695[1]
Core average void fraction	%	70
Core pressure drop	MPa	0.04
Core average Puf content	%	10.4
Puf conversion ratio	—	1.05
Void reactivity coefficient	10^{-4} Δk/k／%void	-0.5
Fuel cycle length	month	24

1: In addition, upper and lower blankets of 0.22 and 0.19 m

3. INVESTIGATION ON CORE CHARACTERISTICS FOR PU MULTIPLE RECYCLING

3.1. Core performances under advanced reprocessing with low DFs

As mentioned in Introduction, the core performances under the advanced fuel reprocessing schemes proposed for the FBR fuel cycle have been investigated. The results of the investigation are overviewed in the following. Advanced fuel reprocessing schemes, such as the advanced PUREX processes and the dry reprocessing ones, have been proposed to give much lower DFs than in the current PUREX process with the very high DFs around 10^6 or more, and hence, some amount of FPs and MAs are contained even in the fresh fuel for the reactor [5]. Since they are expected to have negative effects on the core performances, such as the reactivity, the void reactivity coefficient and the conversion ratio, the effects should be evaluated, considering them as the components of the fresh fuel under the Pu recycling situation. A standard evaluation of the core characteristics, as presented in Table I, was performed for the fuel components obtained under the current PUREX reprocessing of the spent fuel from LWRs. That is, the amount of FPs and MAs in the fresh fuel is negligible in this case. This fuel composition is refered the standard fuel composition in our study on RMWR.

3.1.1. Investigation of effects of MAs on core performances

At first, the effects of FPs or MAs have been investigated individually as the parameter effect study. On the effects of MAs, the evaluation conditions are determined to assume all MAs be contained in the fresh fuel and all FPs be removed from the spent fuel of a typical BWR core with the discharge burn-up of 45 GWd/t. The resultant TRU composition in the fresh fuel is listed in Table II in comparison with the standard fuel composition of our study. The total amount of MAs is 11 wt% of TRU. Especially in this case, the amount of ^{237}Np is large due to production from ^{235}U in the UO_2 core. Although ^{241}Am is included in the standard case, this comes from decay of ^{241}Pu.

In general, the main effect of MAs and FPs in the fuel is to reduce the criticality of the core due to their neutron absorption. In addition, MAs tend to have some effect on the void reactivity coefficient to make it positive. For the present MA effect case, the amount of fissile Pu is to be increased by about 10 % to keep the criticality. Also, MAs, especially ^{237}Np, make the void reactivity coefficient to be positive. Therefore, we have to change the design to keep the void reactivity coefficient to be negative. In the present design, the void reactivity coefficient can be controlled mainly by the length of the upper blanket. If the upper blanket is shortened, the void reactivity coefficient becomes lower. However, it should be noted that this, in turn, makes the conversion ratio lower. For the present case, the upper blanket should be significantly shortened up to about 30% of the standard case. The lengths of the lower and the internal blanket are simultaneously adjusted to 140 and 400mm, respectively, to improve the core performances. As a result, the fissile Pu conversion ratio is reduced to 1.02. This result shows that the RMWR core can be feasible under the MA effect case fuel composition described above, although the effects of MAs are significant. The major dimensions and characteristics of the core are summarized in Table III in comparison with the standard case. In this series of investigation, another and the first core design of 1,100 MWe was used. Its core concept and core performances are close to those for the 1,356 MWe core, and the difference between them are a little except for the burn-up.

Table II. TRU composition for MA effect study case

Item		MA effect case	Standard case
Conditions	Origin core	BWR core (UO_2)	BWR core (UO_2)
	Discharge burn-up (GWd/t)	45	45
	Cooling time for reprocessing (y)	5	5
	Period after reprocessing (y)	2	2
TRU composition (wt%)	^{237}Np	5.6	0.0
	^{238}Pu	2.4	2.7
	^{239}Pu	42.9	47.9
	^{240}Pu	27.2	30.3
	^{241}Pu	8.6	9.6
	^{242}Pu	7.6	8.5
	^{241}Am	3.9	1.0
	^{242m}Am	0.1	0.0
	^{243}Am	1.3	0.0
	^{244}Cm	0.4	0.0
	^{245}Cm	0.0	0.0
	Total	100.0	100.0

Table III. Major core dimensions and characteristics for MA effect case

Item		MA effect case	Standard case
Electric power output	(MWe)	1,100	1,100
Discharge burn-up for core part	(GWd/t)	45	45
Core height	(m)	0.835	0.68
Core average void fraction	(%)	70	70
Core average fissile Pu content	(%)	9.4	10.2
Loaded fissile Pu	(t)	13.8	12.1
Fissile Pu conversion ratio	(－)	1.02	1.06
Void reactivity coefficient	($10^{-4}\Delta k/k$／%void)	-0.5	-1
Operation cycle length	(EFPM)	14	14
Core axial fissile Pu enrichment distribution	mm	DU 100	DU 330
	wt% mm	18 210	18 185
	mm	DU 400	DU 295
	wt% mm	18 225	18 200
	mm	DU 140	DU 200
Amount of MA/FP in MOX (wt%)		3.9 / 0	0.3 / 0

3.1.2. Investigation of effects of FPs on core performances

On the effects of FPs, the evaluation conditions are determined to assume a part of FPs be contained in the fresh fuel based on the concerned DFs and all MAs be removed from the spent fuel of a typical BWR core with the discharge burn-up of 45 GWd/t. Two sets of DFs are selected for the investigation, assuming an advanced PUREX reprocessing and a dry reprocessing scheme [5]. Although the average value of DFs for the latter is much lower and about 10, the same fissile Pu conversion ratio of 1.06 has been attained under the negative void reactivity coefficient. In these cases, the fissile plutonium is also to be increased by about 10 % to keep the criticality.

3.1.3. Investigation on core performances under multiple recycling situation

For more realistic investigation on the multiple recycling situation, their effects have been investigated together as the accompanied FPs and MAs under the multiple recycling with the RMWR and an advanced reprocessing in relatively low DFs. The evaluation conditions are determined to assume all MAs and a part of FPs be contained in the fresh fuel. The average DF of 10 for FPs is adopted assuming a dry reprocessing case [5]. The equilibrium for the recycling has been obtained after about 20 times recycling calculations. The resultant TRU composition in the fresh fuel is listed in Table IV in comparison with the standard fuel composition. The total amount of MAs is 6 wt% of TRU. In this case, the amount of ^{237}Np is much smaller than in the case of Table II due to the MOX core reprocessing. In Table IV, some differences in Pu composition from the standard case are also observed. That is, the rate of ^{241}Pu is decreased and ^{240}Pu is increased in the multiple recycling case.

Table IV. TRU composition for multiple recycling case

Item		Recycling case	Standard case
Conditions	Origin core	RMWR core	BWR core (UO_2)
	Discharge burn-up (GWd/t)	45	45
TRU Composition (wt%)	^{237}Np	0.5	0.0
	^{238}Pu	2.4	2.7
	^{239}Pu	50.6	47.9
	^{240}Pu	34.0	30.3
	^{241}Pu	4.1	9.6
	^{242}Pu	3.2	8.5
	^{241}Am	3.6	1.0
	^{242m}Am	0.1	0.0
	^{243}Am	0.9	0.0
	^{244}Cm	0.5	0.0
	^{245}Cm	0.1	0.0
	Total	100.0	100.0

For the multiple recycling case, the amounts of MAs and FPs contained in the MOX are 1.8 and 1.0 wt%, respectively. Also, the rate of ^{241}Pu, which has good effect on criticality, is decreased. Therefore, fissile plutonium is to be increased by about 10% to keep the criticality. On the void reactivity coefficient, there are about 2wt% of MAs contained in the MOX. In addition, the rate of ^{240}Pu, which has the unfavorable effect on the void reactivity coefficient, is increased as mentioned above. Therefore, we have to change the design to keep the void reactivity coefficient to be negative. As in the MA effect study, we can make the void reactivity coefficient lower by mainly shortening the length of the upper blanket. For the present case, the upper blanket should be significantly shortened up to about 40% of the standard case. The lengths of the lower and the internal blankets are simultaneously adjusted to 150 and 400 mm, respectively, to improve the conversion ratio and the void reactivity coefficient. As a result, the fissile Pu conversion ratio is reduced to 1.02. This result, however, shows that the RMWR core can be feasible under the multiple recycling fuel composition. The major dimensions and characteristics of the core are summarized in Table V in comparison with the standard case.

Based on these investigation, it has been confirmed that the high conversion ratio more than 1.0 and the negative void reactivity coefficient are able to be achieved in the RMWR core by slightly adjusting the basic core design, even under the multiple recycling through the advanced fuel reprocessing schemes with the lower DFs around 10 assumed for FBR fuel cycle. Through the study, the unfavorable effect of MAs on the void reactivity coefficient has been found to be significant. Therefore, in the Pu multiple recycling fuel cycle for the RMWR, a reprocessing scheme with higher DFs for MAs is considered to be favorable.

Table V. Major core dimensions and characteristics for multiple recycling case

Item		Recycling case		Standard case	
Electric power output	(MWe)	1,100		1,100	
Discharge burn-up for core part	(GWd/t)	45		45	
Core height	(m)	0.84		0.68	
Core average void fraction	(%)	68		70	
Core average fissile Pu content	(%)	9.4		10.2	
Loaded fissile Pu	(t)	13.7		12.1	
Fissile Pu conversion ratio	(−)	1.02		1.06	
Void reactivity coefficient	($10^{-4}\Delta k/k$／%void)	-0.5		-1	
Operation cycle length	(EFPM)	14		14	
Core axial fissile Pu enrichment distribution	mm	DU	130	DU	330
	wt% mm	18	215	18	185
	mm	DU	400	DU	295
	wt% mm	18	225	18	200
	mm	DU	150	DU	200
Amount of MA/FP in MOX (wt%)		1.8 / 1.0		0.3 / 0	

3.2. Core performances under advanced reprocessing with relatively high DFs

In JAERI, another reprocessing scheme is proposed for the RMWR fuel cycle based on its own reprocessing study [6]. This process, named innovative PUREX, is a kind of simplified PUREX process and is expected to be economical by eliminating the purification steps for Pu and U as shown in Figure 3. Although the purification steps are eliminated, a new chemical treatment is introduced prior to the U/Pu partitioning step and eliminates Np and FPs. The resultant average DF for FPs is about 10^5 comparing with about 10^7 for the PUREX and DF for Np is expected to be about 100 as in the PUREX. Another favourable point for this process is that the average DF is high enough to utilize the existing fuel fabrication process in the globe boxes. Therefore, this process is expected to be advantageous for early establishment of the fuel cycle based on the existing technologies.

For the multiple recycling case under this reprocessing process has also investigated. The resultant TRU composition in the fresh fuel is listed in Table VI in comparison with the standard fuel composition. The total amount of MAs is 0.2 wt% of TRU. In this case, the amount of ^{237}Np is much smaller than in the case of Table 4 due to the special chemical treatment in this process. In Table VI, some differences in Pu composition from the standard case are also observed. That is, the rate of ^{238}Pu, ^{241}Pu, ^{242}Pu are decreased and ^{239}Pu, ^{240}Pu are increased in the multiple recycling case. This trend is also observed in Table IV.

For this multiple recycling case, the amount of MAs and FPs contained in the MOX is negligibly small, but the rate of ^{241}Pu, which has good effect on criticality, is decreased. Therefore, fissile plutonium is to be increased by about 5 % to keep the criticality, resulting in increase of MOX length to 20.5cm. The effects on the major core performances such as the conversion ratio, the void reactivity coefficient, the burn-up are not noticeable degree.

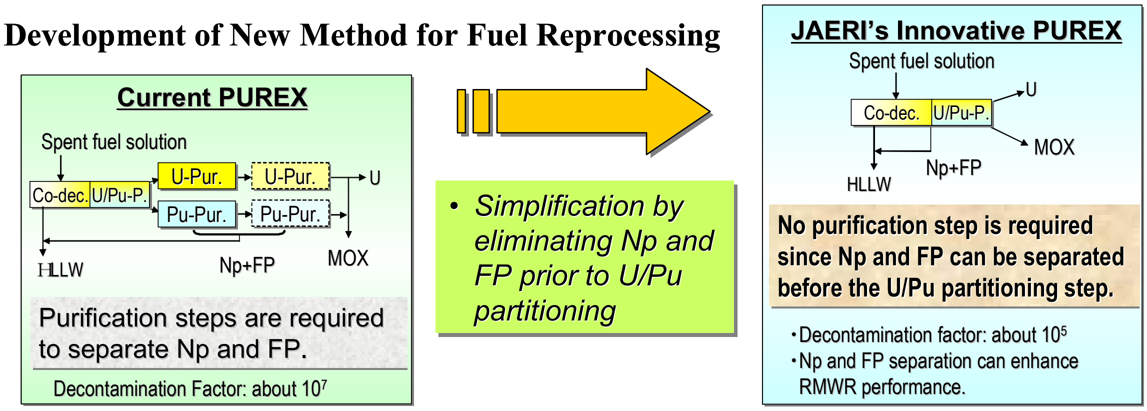

FIG. 3. JAERI's innovative PUREX reprocessing process proposed for RMWR fuel cycle.

4. OTHER DETAILED INVESTIGATION ON CORE DESIGN

As shown in Figure 2, the present core design has unique double-flat-core type one and the MOX region is axially divided into two regions by the internal blanket. Also, the core power is controlled only by the control rod. Therefore, detailed analyses for the control rod operation plan have been performed to establish a favorable plan without problems such as a strong power peaking.

Table VI. TRU composition for multiple recycling case under innovative PUREX process

Item		Recycling case	Standard case
Conditions	Origin core	RMWR core	BWR core (UO_2)
	Discharge burn-up (GWd/t)	45	45
TRU Composition (wt%)	^{237}Np	0.0	0.0
	^{238}Pu	0.6	2.7
	^{239}Pu	55.7	47.9
	^{240}Pu	36.2	30.3
	^{241}Pu	4.5	9.6
	^{242}Pu	2.8	8.5
	^{241}Am	0.2	1.0
	^{242m}Am	0.0	0.0
	^{243}Am	0.0	0.0
	^{244}Cm	0.0	0.0
	^{245}Cm	0.0	0.0
	Total	100.0	100.0

In the investigation, 61 rods are assigned to be operated during the full power operation cycle out of the total 283 control rods. The rest of 222 rods are assigned to be withdrawn during the start-up operation, in some groups. At the beginning of the cycle (BOC), above mentioned 61 control rods are inserted up to around the upper MOX region. They are gradually withdrawn through the cycle and are completely withdrawn at the end of the cycle (EOC). The resultant core average axial power distribution is presented in Figure 4 at BOC and EOC. The relative axial power peaking is around 1.6 at maximum and this value is not changed throughout the cycle. On the other hand, the radial peaking is around 1.2 at maximum. Therefore, it has been confirmed that the core power can be adequately controlled by the control rod operation.

Furthermore, other studies on the RMWR core, such as core performance improvements, start-up sequence planning, initial core design and so forth has been performed.

5. INVESTIGATION ON THERMAL HYDRAULICS IN TIGHT-LATTICE CORE

From the thermal hydraulic point of view, such a tight- lattice rod configuration as presented above may result in some difficulties. One of the major issues to be concerned for the tight-lattice core is the critical heat flux (CHF) under the relatively small gap width of about 1 mm. Although there are some unclassified experimental data on CHF for the tight-lattice core, they do not cover the range concerned in the present RMWR design. Therefore, the investigation on CHF in the tight-lattice core has been performed in JAERI, including experimental activities [7]. The purposes of the investigation are (1) to make clear the parameter effects on critical power, (2) to evaluate the existing design correlation, (3) to develop the high accuracy correlation, if necessary and (4) to verify the detailed numerical analysis code. The left hand side of Figure 5 shows a schematic of the 7-rod bundle CHF test section. It simulates the double-flat-core type axial power distribution in the RMWR core. The rod diameter and gap width between rods are 13mm and 1.3mm, respectively. The test section is installed into the high pressure water circulating loop simulating the flow conditions expected in the RMWR operation.

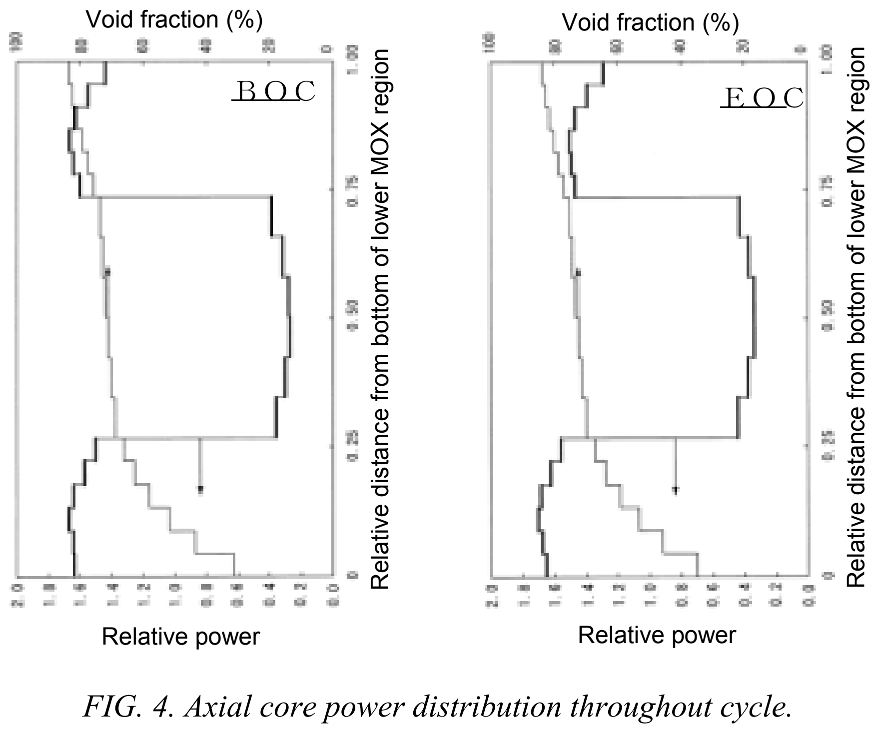

FIG. 4. Axial core power distribution throughout cycle.

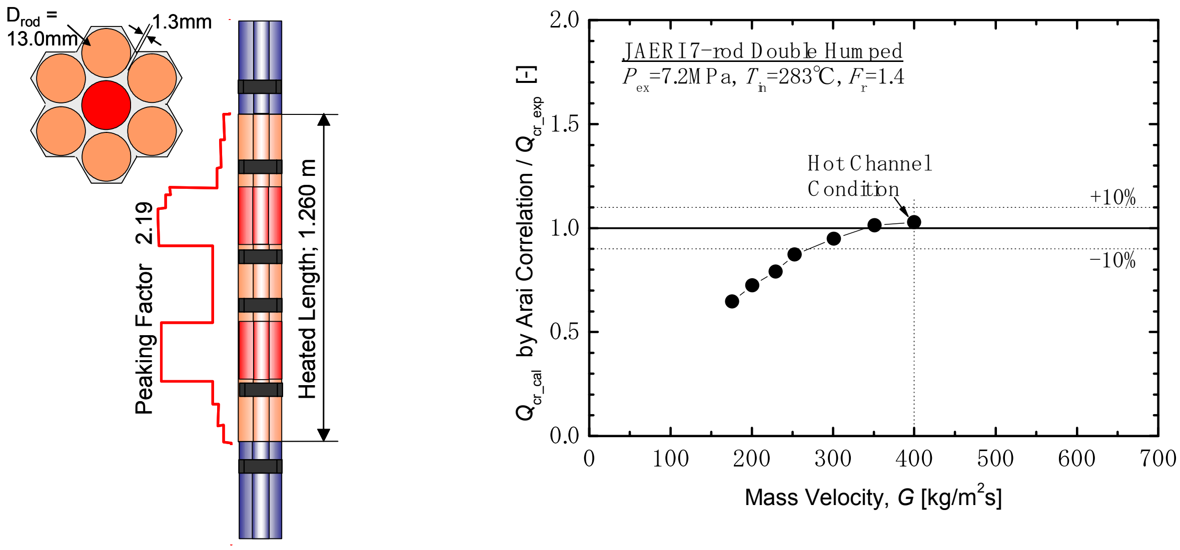

FIG. 5. Schematic of 7-rod bundle CHF test section and comparison between CHF data and prediction.

Some of the CHF data are compared also in Figure 5 with the estimated values with the CHF correlation [8] used in the core design. In the figure, the ratios of the predicted CHF with the correlation to the measured are presented for the expected mass velocity range. The results indicate that the predicted values with the correlation are almost the same or smaller than the experimental data. Therefore, the correlation is confirmed to be adequate from the safety point of view to predict the CHF in the tight-lattice core.

At present, a new correlation is under development, because the discrepancy is large in the low mass velocity region. Furthermore, larger scale thermal hydraulic experiments using 37-rod test section are planned for confirmation of the scale effects of the test section. Using the present CHF data as well as the additional data obtained by the fundamental model experiments, thermal hydraulic design codes including the transient analysis and the subchannel analysis codes have been optimized for the tight-lattice core. Safety analyses for the major abnormal transients and accidents have shown its enough safety margin [9].

6. CONCLUSIONS

An innovative water-cooled reactor concept named RMWR is under development by JAERI, aiming at effective fuel utilization through plutonium multiple recycling based on the well-experienced water reactor technology. RMWR is able to achieve a high conversion ratio more than 1.0 and the negative void reactivity coefficient. Up to the present, the representative RMWR core design is selected based on the BWR-type concept and has been been investigated in detail.

Especially from the multiple recycling point of view, investigation on the RMWR core performances under multiple recycling situation with the advanced reprocessing schemes with relatively low DFs has been performed. Through the investigation, it has been confirmed that the high conversion ratio more than 1.0 and the negative void reactivity coefficient are able to be achieved in the RMWR core by slightly adjusting the basic core design, even under the multiple recycling including the advanced fuel reprocessing schemes with the low DFs. In addition, it has been confirmed that the innovative PUREX process proposed by JAERI is favourable one for the RMWR fuel cycle.

Through other detailed investigation on the core design, detailed control rod operation plan has been established without any serious power peaking even in the special double-flat-core type design. Thermal hydraulic investigation in the tight-lattice core has also been performed including the experimental activities and they indicate promising results.

7. REFERENCES

[1] IWAMURA, T., *et al*., "Development of Reduced-Moderation Water Reactor (RMWR) for Sustainable Energy Supply", *Proc. 13 th Pacific Basin Nuclear Conf. (PBNC2002)*, TS-8C:6 (2002)..

[2] OKUBO, T., *et al*., "Conceptual designing of Reduced-Moderation Water Reactors (1) – Design for BWR-type reactors --", *Proc. ICONE 8*, ICONE-8422 (2000).

[3] HIBI, K., *et al*., "Conceptual designing of Reduced-Moderation Water Reactors (2) – Design for PWR-type reactors --", *Proc. ICONE 8*, ICONE-8423 (2000).

[4] OKUBO, T., *et al*., "Design of Small Reduced-Moderation Water Reactor (RMWR) with Natural Circulation Cooling", *Proc. PHYSOR 2002*, 5B-4 (2002).

[5] IKEGAMI, T., *et al*., "Study of Core Characteristics on Fuel and Coolant Type – Results of F/S Phase-1 --", TY9400 2001-011 (2001), [in Japanese].

[6] UCHIYAMA, G., *et al.*, "PARC process for Reprocessing Spent Fuels from Advanced Nuclear Reactors", *Proc. 13 th Pacific Basin Nuclear Conf. (PBNC2002)*, TS-7C:7 (2002).

[7] LIU, W., *et al*., "Critical Power Characheristics of Tight Lattice Bundles", *Proc. ICONE-11, ICONE11-36099* (2003).

[8] ARAI, K., *et al*., "Critical Power Characteristics of a High Conversion Boiling Water Reactor", *IAEA Technical Committee on Technical and Economic Aspects of High Converters* (1990).

[9] OKUBO, T., *et al.*, "Study on Safety and Core Improvement of Reduced-Moderation Water Reactor (RMWR) with High Conversion Ratio", JAERI-Research 2001-021 (2001), [in Japanese].

STATE OF THE ART OF SECOND PHASE FEASIBILITY STUDY ON COMMERCIALIZED FAST REACTOR CYCLE SYSTEM

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Abstract. Japan is preceding the development of the fast reactor (FR) and related fuel cycle system as a priority target to secure its own energy supply. In addition, the commercialization of FR cycle system can be expected to contribute greatly to security of energy resource and preservation of global environment in the world. To deliver this attractive feature, the feasibility study on commercialized fast reactor cycle system (Feasibility Study) was launched in 1999 by the organization composed of Japan Nuclear Fuel Cycle Development Institute (JNC), Japanese electric utilities, Central Research Institute of Electric Power Industry (CRIEPI) and Japan Atomic Energy Research Institute (JAERI). The Feasibility Study aims to present the prominent FR cycle technologies for commercialization by around 2015. In consideration of future social needs, five development targets (ensuring safety, economic competitiveness, reduction in environmental burden, efficient utilization of resources and enhancement of nuclear non-proliferation) were set up at the start point. The first phase Feasibility Study was carried out in the period from 1999 to 2000 and the second phase launched in April 2001. The period of the second phase study is for five years. Feasible candidates screened out in the first phase study are being investigated with the adoption of innovative technologies to clarify promising candidates. In this paper, the evaluation method for the clarification of the promising candidates is briefly discussed, and the progress of the design study and experimental tests for key technologies is presented.

1. INTRODUCTION

In the 20th century, civilization developed drastically by rapid evolution of science and technology. Although peoples of advanced countries have enjoyed materially rich life, the fears of energy resource depletion and environmental destruction have been gradually actualized. In addition, it is expected that the total population in the world amounts to about 10 billion until the middle of the 21st century. As for the energy demand, it is possible along with this increase in population to increase by about 1.5 to 2 times more than at present.

Supposing this energy need is boarded with fossil fuels such as oil and natural gas, it worries not only the shortage of fossil resources, but also the influence on global warming caused by the exhaust of carbon dioxide. These have adverse impacts on the living conditions. Moreover, focusing on developing countries, the growth of large energy demand is expected and the global scale aggravation to energy and environmental problems has been worried in the 21st century.

Since Japan is very poor in energy resources, it is necessary to save them to supply with stability for a long period. Furthermore, the technology development with small environmental burden is indispensable in energy production. To deal with the problems on energy supply and demand in the 21st century, Japan is promoting the improvement of energy efficiency and the development of renewable energy technologies.

As for nuclear energy, the deployments of new reactor plants are stiffened due to the anxiety to safety and the uncertainty of radioactive waste disposal. On the other hand, the necessity of nuclear power is reviewed under the backgrounds of the recent steady operation records of nuclear plants, the necessity of energy supply with the global environmental preservation, and the progress of the disposal business, etc. In the international corporations of GEN-IV and INPRO, the research and development of nuclear power system including fuel cycle for next generation are in progress.

Nuclear power has no greenhouse gas emission and the amount of waste for a unit energy generation is extremely little. The spent fuel recycle by the introduction of reprocessing is one of the promising candidates for sustainable energy sources of the 21st century. Especially, the effective use of uranium resource is spectacularly improved by the fast reactor (FR) cycle. The energy supply by nuclear power becomes possible for a long term of hundreds of years or more, therefore the FR cycle is expected as a future nuclear power system.

As Japan is very poor in natural resources and imports 80% of them, it is very week in domestic energy supply and is not good to depend on the only energy source. Electric power companies in Japan seek the optimal combination of power sources as shown in Figure 1. Power companies currently use nuclear power as their base load supply, because it is better for economic performance and supply stability. Especially, the cost of oil and natural gas is relatively higher compare to other countries, thus the price competitiveness of nuclear power is higher in Japan. The optimum mixture of energy supply is important so that the government supports the development of various kinds of energy sources, such as hydro power, oil, natural gas, nuclear power, window power, photovoltaic power, and so on.

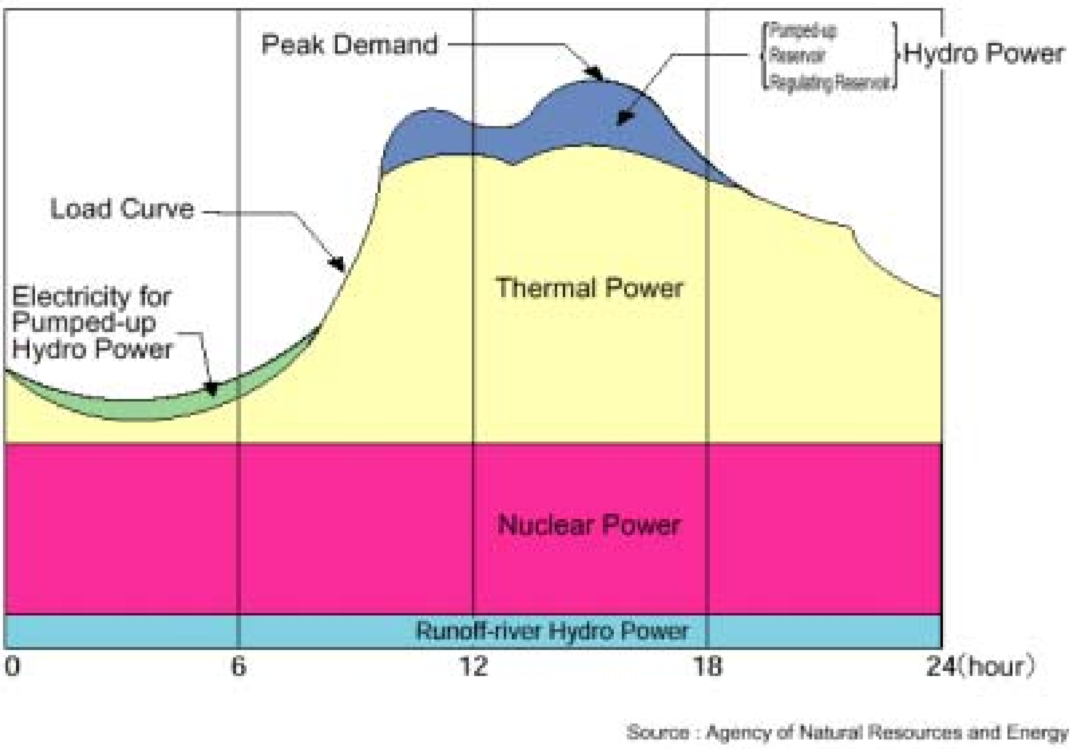

FIG. 1. Optimal combination of power sources in Japan.

In addition to economic performance and supply stability, nuclear power plants emit much lower carbon dioxide than that of solar and wind power, as well as thermal power of fossil energy sources, such as oil, natural gas and coal. The Kyoto Protocol on curbing global warming (COP3) determined the legal binding reduction target to advanced countries and it showed to reduce about 5% of greenhouse effect gases in all advanced countries. In order to stabilize the amount of greenhouse gas emission, increase of nuclear power plant and introduction of non-fossil energy and countermeasure of energy saving are necessary. There are currently 51 nuclear power plants in operation in Japan, producing 34% of electricity. As for the renewable non-fossil energy, Japan is a leading country in the world of photovoltaic power generation and wind power generation capacity has been rapidly increasing for the last several years.

2. FEASIBILITY STUDY ON COMMERCIALIZED FR CYCLE SYSTEM

The FR cycle is important as the technology that achieves long-term use for nuclear power aiming at resource saving and little waste by recycling spent fuels. In Japan, realization of the FR and related fuel cycle system is a priority target to secure its own energy supply. In addition, the commercialization of FR cycle system can be expected to contribute greatly to security of energy resource and preservation of global environment in the world. To deliver this attractive feature, the Feasibility Study (FS) on commercialized fast reactor cycle system is being performed by the organization composed of Japan Nuclear Fuel Cycle Development Institute (JNC), Japanese electric utilities, Central Research Institute of Electric Power Industry (CRIEPI) and Japan Atomic Energy Research Institute (JAERI).

Figure 2 depicts the development steps of FS for FR cycle systems. FS aims to present the prominent FR cycle technologies for commercialization by around 2015. The first phase FS was carried out in the period from 1999 to 2000. A concept of innovative recycle system that can efficiently reprocess and fabricate TRUs, and burn them in the FR is studied. As a results of the first phase in about two years, several promising FR plants and the related fuel cycle systems have emerged as candidates for the future FR cycle system which can attain the development targets for commercialization.

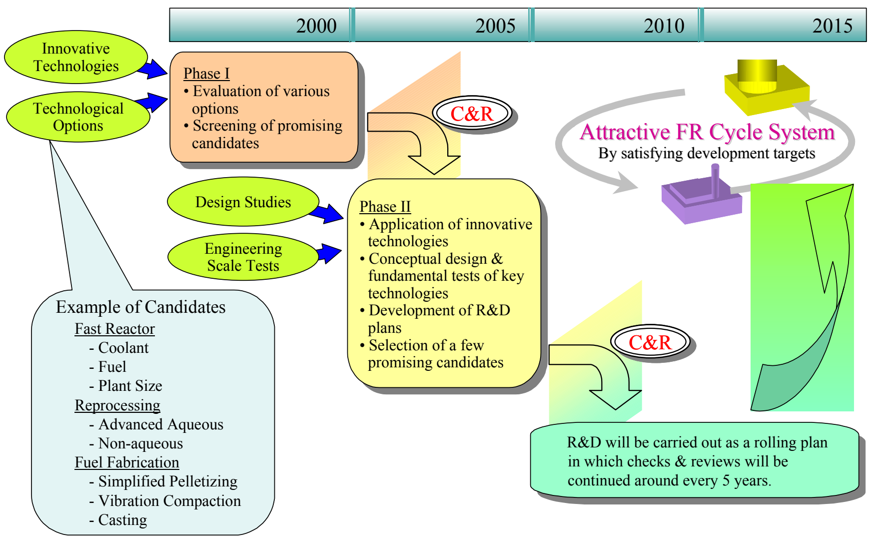

FIG. 2. Development steps of feasibility study for FR cycle system.

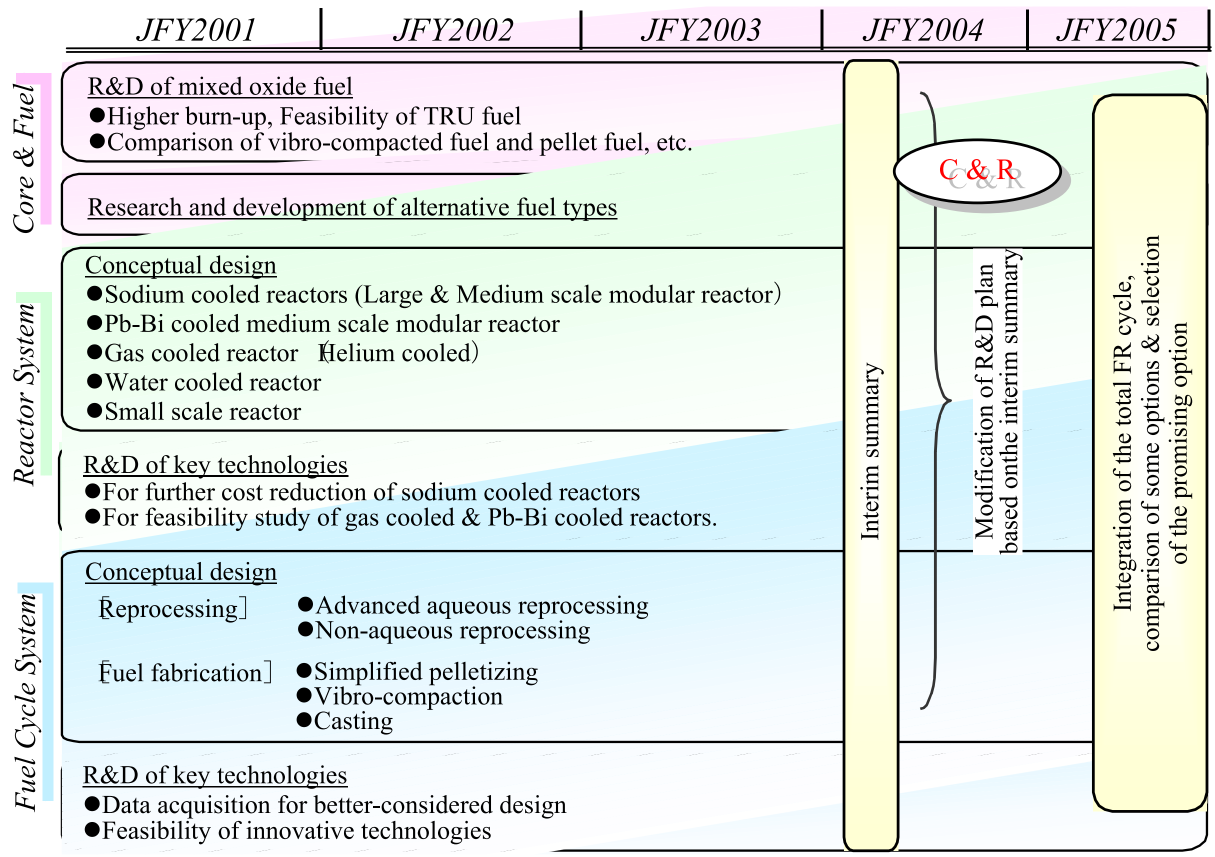

FIG. 3. Scope of the second phase feasibility study.

Following the first phase FS, the second phase launched in April, 2001 as shown in Figure 3. The period of the second phase FS is for five years. Feasible candidates screened out in the first phase are being investigated with the adoption of innovative technologies [1-7]. In the second phase FS, the large variety of design study and experimental tests of key technologies for each candidate are being conducted to confirm the feasibility of candidates. The interim report of the second phase FS will be drawn up in JFY 2003 and the framework of promising candidates and related roadmaps will be indicated in consideration of the technical consistency between FR and fuel cycle system.

International cooperation is important and effective as well as domestic collaboration in developing the FR cycle system. FS is being conducted in cooperation with the related research organization in Europe, U.S.A. and Russia as shown in Figure 4. The information exchange concerning the design and fuel development for helium gas-cooled FR has been being well achieved between CEA and JNC. As for lead-bismuth-cooled FR, the fundamental experiments of corrosion/erosion characteristics against structure are underway by a collaborative work with FZK/KALLA. The EAGLE project, which is aiming at demonstrating the effectiveness of inner duct structure to enhance the fuel discharge without propagating neighbor subassemblies and to obtain a perspective for re-criticality free concept in sodium-cooled MOX fuel core, is in progress between .NNC (National Nuclear Center)/RK (Republic of Kazakhstan) and JNC with Utilities. In addition, Japan is nominated as leading country for the development of GEN-IV SFR (Sodium-Cooled Fast Reactor System) [8].

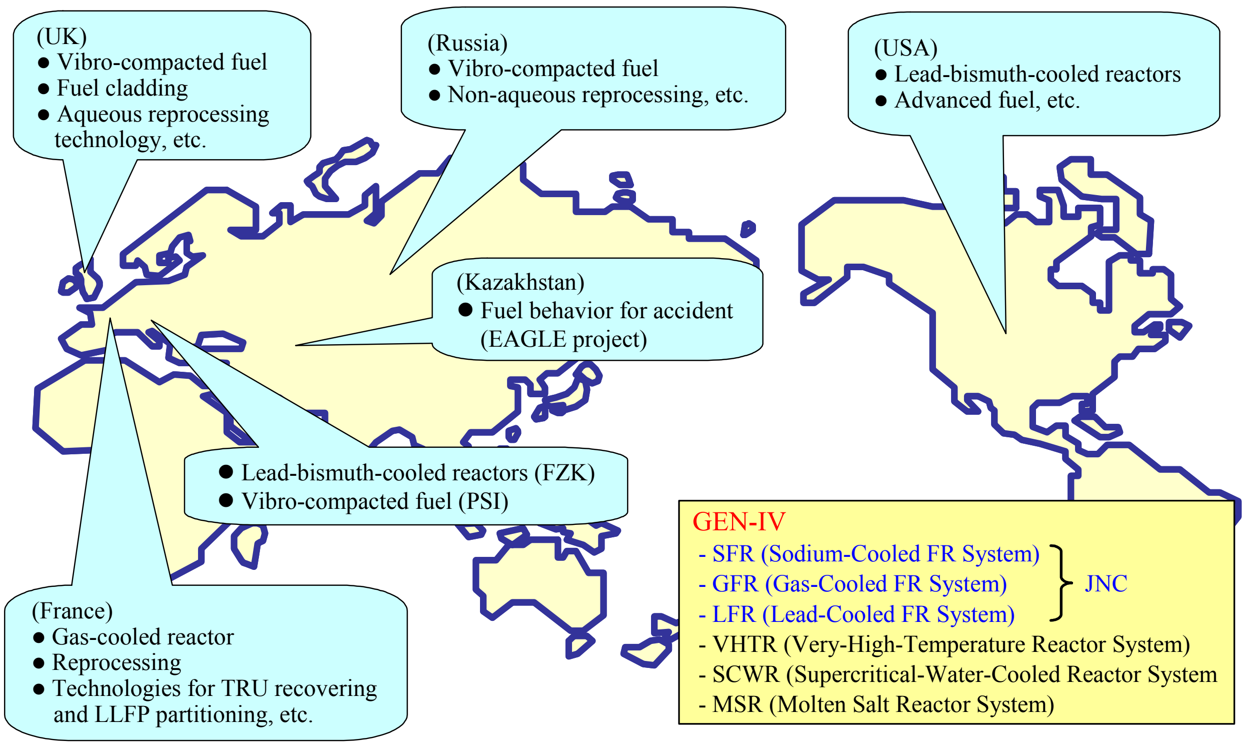

FIG. 4. International cooperation.

In this paper, the progress of the design study and experimental tests for key technologies will be presented, and the evaluation procedure for the promising candidates will be briefly discussed.

3. TARGETS FOR FEASIBILITY STUDY

On the major premise of safety, it is expected to the energy system of the 21st century to be able to reduce the environmental burden and to supply energy for the long term. The improvement of economy is indispensable to be built upon business under electric power trade liberalization. In addition, nuclear proliferation resistance is requested from the international aspect to prevent nuclear weapon diffusion. Under such circumstances, the FR cycle technology excels in neutron economy, high energy supply ability, excellent TRU (transuranium) burning and LLFP (long-lived fission product) transmuting characteristics etc., and has potential that satisfies the above-mentioned requirements.

In consideration of future social needs, five development targets (ensuring safety, economic competitiveness, reduction in environmental burden, efficient utilization of resources and enhancement of nuclear non-proliferation) were set up at the start point of FS. Needless to say, ensuring safety is the basic premise in the development of FR cycle system. Economic competitiveness, as today's most important issue, has to be drastically improved by the development of innovative technologies for the implementation of FR cycle system. The radioactive waste generation, as one of the crucial issues of fission energy use, has to be improved in the process for the handling of long-lived radiotoxicity. The contents of the five development targets are described below and summarized in Figure 5.

- Safety
  - Risks caused by introduction of FR < Risks that already exist in society
- Economic Competitiveness
  - Power generation cost ≦ Future LWRs, other energy resources
  - To ensure cost competitiveness in the global market
- Reduction of Environmental Burden
  - To reduce impact of high level radioactive waste of repository by means of burning or transmuting long-life nuclides (TRUs and LLFPs)
  - To reduce radioactive waste generated in the course of plant operation and maintenance as well as decommissioning
- Efficient Utilization of Natural Resources
  - To produce sustainable nuclear fuel through applying the advantage of the neutron economy
  - To respond to diverse needs for energy resources (Production of hydrogen, desalination of sea water, heat supply, dispersion of power sources, etc.)
- Enhancement of Nuclear Non-proliferation
  - To reduce burden of nuclear physical protection and safeguards (Isolate pure plutonium in any FR cycle process)
  - To effectively operate non-proliferation system (Remote monitoring system, etc.)

FIG. G. Development targets of commercialized FR cycle systems.

3.1. Ensuring safety

Safety is a major premise for not only nuclear systems but also any engineering systems to be accepted in the society. In the development and operation of FR cycle system, it is essential to examine how to ensure the safety of every facility in its design, construction, operation and decommissioning stages, keeping in mind potential risk due to the existence of a great deal of nuclear fuels and radioactive materials inside.

We adopt the safety design that has basic philosophy of defense-in-depth and gives top priority to the prevention of occurrence and expansion of abnormalities. With this design framework, we keep the safety at higher level than or equivalent to that of the light water reactor of the same generation.

Furthermore, based on the characteristics of each facility, we aim at building more certain and clearer safety measures. For this purpose, a passive safety mechanism is installed or strengthened to prevent core disruptions. In the event of a hypothetical core disruption, re-criticality should be avoided and the event should be terminated inside the reactor vessel or the containment vessel. Moreover, the safety design takes into consideration physical and chemical characteristics (chemical activity, radiotoxicity etc.) of materials handled in a nuclear reactor plant or a fuel cycle facility.

The safety design aims at keeping the risk of introducing FR cycle system small enough comparing to the risks already existing in the society.

3.2. Economic competitiveness

For commercial use of the FR cycle system, it is essential to achieve the economy that allows the introduction of the system based on the principle of market mechanism. For this reason, the target on this issue is to have a competitive edge in power generation cost over future competing energy sources. Furthermore, aiming at world-class cost competitiveness, we look to overseas procurement for the further improvement of economics.

3.3. *Reduction of environmental burden*

In the commercial use of the FR cycle system, it is necessary to maximize its attractiveness, such as excellent thermal efficiency, the small amount of waste per energy generation, and the reduction of greenhouse gas emission etc. Taking advantage of favorable neutron economy of the FR cycle system, there is a possibility to reduce exposed dose and risk by the geological disposal of radioactive wastes. Thus, we investigate the separation and transmutation of long-lived radioactive elements (TRU, LLFP) accumulating in fuel cycles to reduce the radioactivity and the potential toxicity of the high level waste. It is also addressed to reduce the volume of radioactive wastes generating in operation, maintenance and decommission of cycle facilities.

3.4. *Efficient utilization of resources*

The long-term energy demand in the world is predicted to increase. However, there are various uncertainties in energy supply and demand predictions. FR has the excellent features of efficient burning of TRU including higher order plutonium isotope as well as favorable neutron economy. These features enable us to use nuclear power as sustainable energy source over the long period of hundreds years or more by recycling uranium resource. The establishment of the FR cycle technology as one of the energy options means to be able to correspond to these uncertainties flexibly. In addition to use FR as base load power supply, we investigate various FR business chances, such as distributed power supply, heat supply, hydrogen production etc.

3.5. *Enhancement of non-proliferation*

In the commercial use of the FR cycle system, we have to openly and clearly show the peaceful use of nuclear energy to the global society, which eliminates the risk of nuclear materials being diverted to nuclear weapons. In order to secure peaceful uses of nuclear materials, nuclear facilities in Japan accept full inspections by the government and safeguards of IAEA based on Non-Proliferation Treaty (NPT). In addition to these extrinsic barriers, the highly radioactive material contaminated fuel provides proliferation resistance feature as an intrinsic barrier. Thus, we investigate a reprocessing system with no pure plutonium in all processes. The remote maintenance, surveillance and fuel fabrication technologies are also important to handle minor actinides and fission products contaminated fuels. The enhancement of intrinsic barrier expects to lead to reducing the correspondence to safeguard.

4. COMPREHENSIVE EVALUATION METHOD

4.1. *Flow of comprehensive evaluation*

Figure 6 shows the flow chart of comprehensive evaluation in FS for various FR cycle concepts. Development targets and design requirements were set up at the starting point, and then design studies and key technology tests are being conducted to meet those targets and requirements. R&D results are summarized in the technical database system for FR cycle to evaluate the achievement levels to the design requirements from technical viewpoint.

Design information on the core, reactor, reprocessing and fuel fabrication systems (which are the units of the design work) is combined to create attractive and/or representative FR cycle concepts. Taking into consideration their rationality as a combination, similarity of cycle characteristics and compatibility between the FR system and the fuel cycle system, candidates of FR cycle concepts are clarified as shown in Figure 6 [9].

The promising candidates of FR cycle system are comprehensively evaluated from viewpoints of compatibility with development targets, technical feasibility and social acceptance. Followed by the evaluation, key FR cycle concepts are clarified, which is developed towards commercial introduction, and then the R&D plan for future FR cycle systems is submitted.

4.2. Evaluation indices

A set of system evaluation criteria corresponding to FS development targets is being studied to provide measures to objectively judge the performance of FR cycle system concepts. Table II depicts seven evaluation indices prepared in FS. In developing the evaluation indices, quantification was considered wherever possible and the evaluation indices were stratified using a multi-layer hierarchy to clearly indicate the relationship between the development targets and the indices.

4.3. Qualitative evaluation method

Considering the characteristics of evaluation indices and design information level available at present, the above seven evaluation indices are classified into three categories. The first category is those indices that can be quantified, including “Economics”, “Resource utilization” and “Environmental impact”. The second category is the indices that are prerequisite for nuclear use, and it is essential to meet the evaluation criteria depending on the system characteristics. “Safety” and “Proliferation resistance” are included in this category. The third category is the indices that are hard to be quantified and needed the judgment based on the expert judgment. This category includes “Technical feasibility” and “Social acceptance”.

For the indices in the first category, hierarchical structure is developed as shown in Figure 7. The utility function U(X), ranging from 0 to 1 of its value, is allocated to each attribute in the lowest layer, and estimates non-dimension worth. Weighting factors are set to attributes so that the total weighting factor in one level should be one, to estimate the attribute in the upper level. The weighting factors are set based on engineering judgment. With the same procedure, we estimate non-dimension worth of the higher-level attribute in the hierarchy. The final non-dimension value means the achievement level to the development target corresponding to this evaluation index.

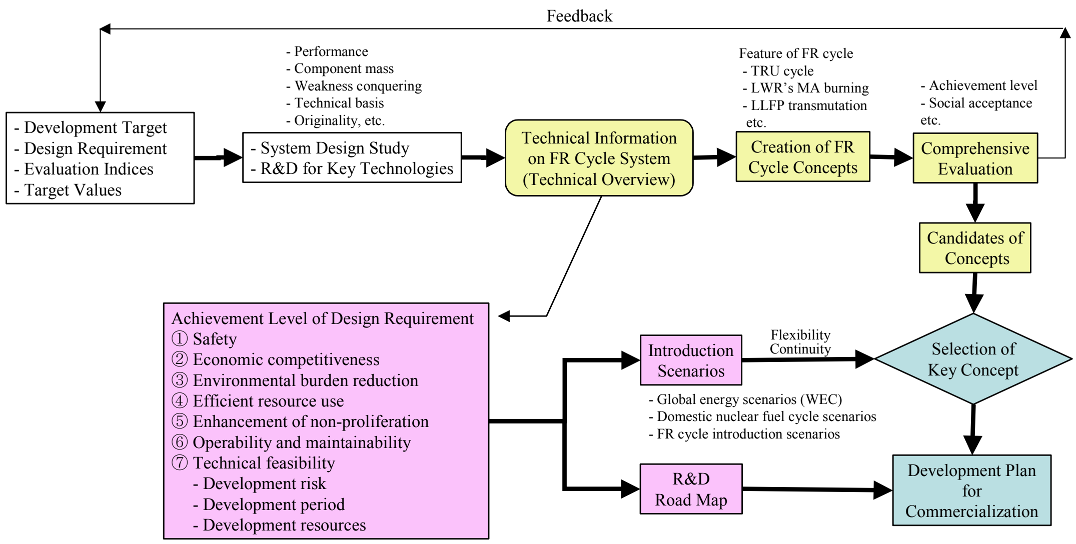

FIG. 6. Schematic flow of comprehensive evaluation.

Table. II. Evaluation indices for the feasibility study.

Development targets	Evaluation indices	First index	Second index	Third index
Economic competitiveness	Economics	Power generation cost (yen/kWh)	- Investment cost - Operation and maintenance cost - Fuel cycle cost, etc.	
Efficient utilization of resources	Resource utilization	Cumulative uranium demand (tonU)	- Out-pile cycle time - TRU inventory, etc.	
		Efficiency of uranium use (%)	- Burn-up - Recover factor, etc.	
Reduction in environmental burden	Environmental impact	Concentrate and retain	- Waste volume	- High level waste - TRU waste - Low level waste, etc.
			- Waste volume converted repository area	
			- Radioactive toxicity	
			- Exposure dose (Repository site)	
		Dilute and disperse	- Exposure dose (Surrounding)	- Ocean - Sea
Ensuring safety	Safety	Design basis events (DBE)	- Guide line for safety design - Safety assessment	
		Margin for beyond DBE	- Prevention of core damage - Elimination of re-criticality - In-vessel retention (PAHR)	
Enhancement of non-proliferation	Proliferation resistance	Intrinsic factors	- Isotopic composition of Pu - Diversion barrier of process, etc.	
		Extrinsic factors	- Safe guard - Physical protection, etc.	
-	Technical feasibility	Development risk	- Technology level	
			- Difficulty level	
		R&D investment	- R&D budget	
			- R&D period	
-	Social acceptance	Benefit	- Responsiveness to social needs - Benefits to local community, etc.	
		Risk	- Scientific risks - Psychological factors, etc.	

The utility function is defined through three points. The allowable performance has a non-dimension value of zero corresponding to the lowest limit for introducing the FR cycle. The sufficient performance of commitment level has a value of 0.5 to smooth introduction of the FR cycle. The challenging target has a highest value of one to promotion or ideal level. We adopted an exponential function with three constants A, B and C as shown in Figure 8 [10]. Figure 9 shows a case of power generation cost, where these three points are specified as follows:

— Allowable limit: 10 yens/kWh corresponding to the renewable wind power
— Commitment level: 4 yens/kWh corresponding to the future FR cycle
— Challenging target: 3 yens/kWh corresponding to the present natural gas thermal power in overseas

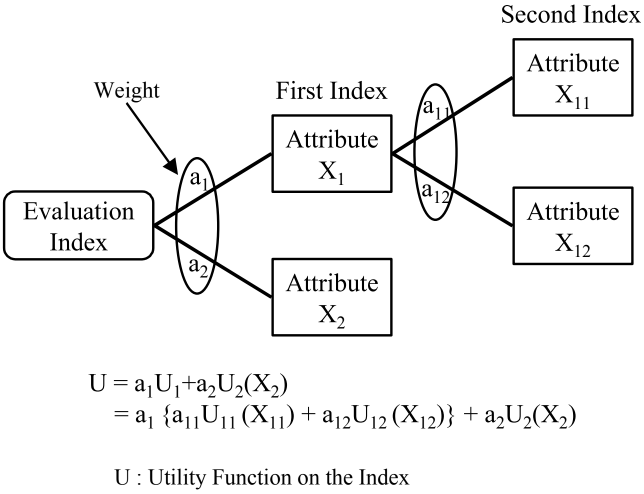

FIG. 7. Hierarchical structure of evaluation index.

As for the “Safety” in the second category, we confirm that each technology meets the judgment standards specified by “Guide line for safety design”, “Safety assessment” and “Margin for beyond design basis events (prevention of core damaged, elimination of re-criticality, in-vessel retention and post accident heart removal) ”. These standards should be a premise to be the subject of FS. As for the “Proliferation resistance”, we prepare a checklist that indicates the evidence of engineering judgment by experts. About the "Technical feasibility" in the third category, we estimate “Development risk (technology level and difficulty level)” and “R&D investment (R&D budget and period)” based on the expert judgment. For the "Social acceptance", we examine the significance and the necessity of the FR cycle, such as externality of environment effect, comparative evaluation with other power sources, social risks, return-on-investment evaluation and introduction scenarios of the FR cycle. We also plan to examine the benefits and risks for individual and community in consideration of psychological factors.

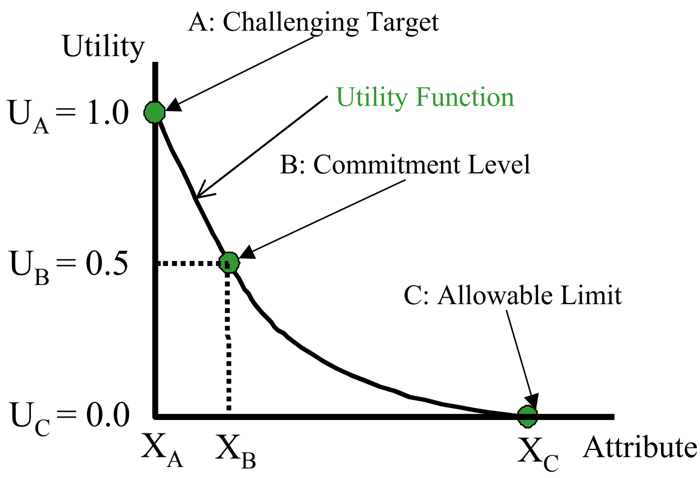

FIG. 8. Utility function.

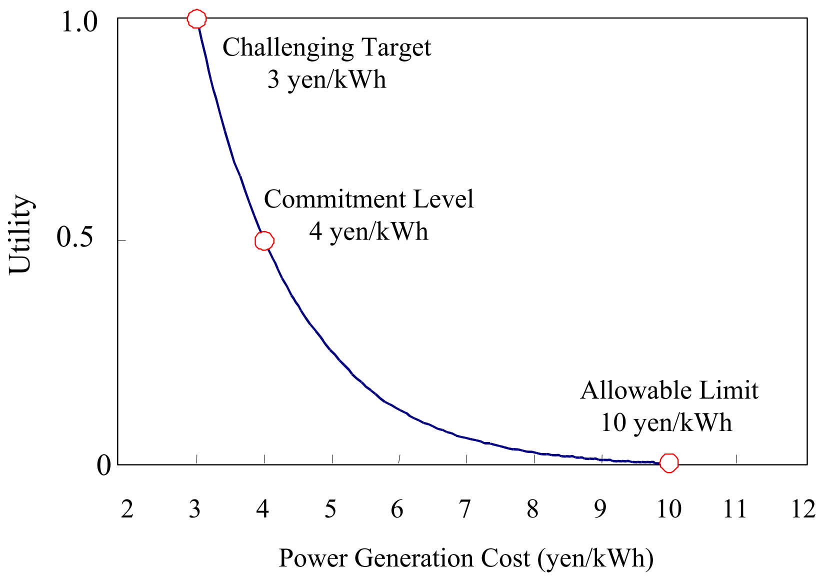

FIG. 9. Utility function of power generation cost.

5. CURRENT STATUS AND FUTURE WORK

5.1. Fast reactor systems

5.1.1. Sodium-cooled fast reactor [11-29]

With regard to system study, the conceptual design of advanced loop-type sodium-cooled FR was confirmed focusing on the improvement of economic competitiveness (Fig. 10). In order to achieve the reduction of the construction cost, the plant design has been investigated from three approaches, such as scale merit, standardization and learning effects, and the design improvement by employing the innovative technologies.

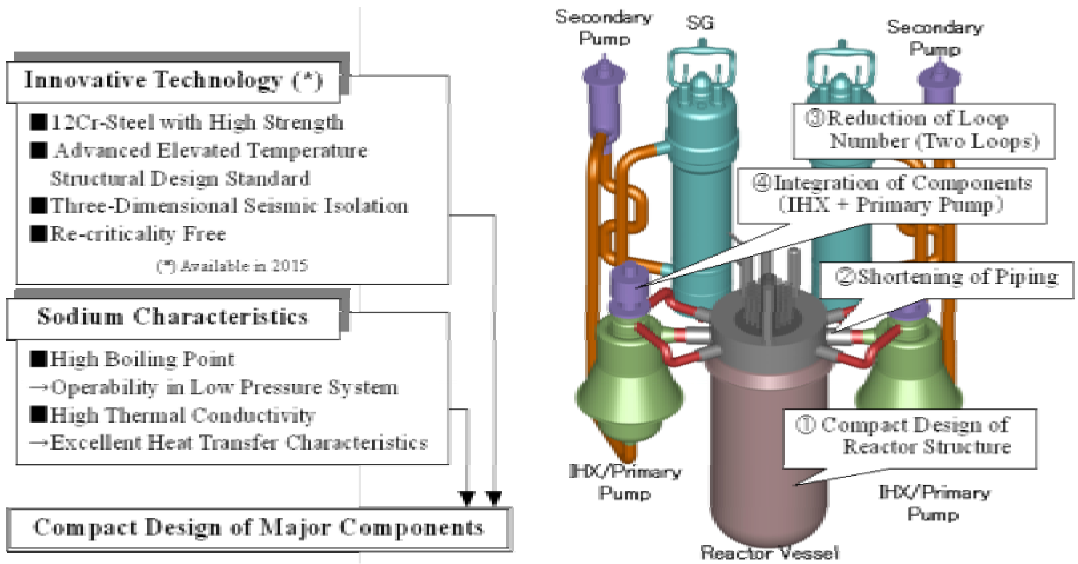

FIG. 10. Design improvements of sodium-cooled fast reactor.

R&D activities for key technologies related to the advanced loop-type reactor concept are: compact reactor vessel (R/V), large diameter piping under high flow velocity condition, and integrated components with a primary pump and an IHX. The compact design of R/V results in high sodium velocity in the upper plenum. Scaled model water tests are being conducted to seek the design measures for stabilizing the coolant flow in the plenum and preventing gas entrainment at the free surface. The two-loops design raised several R&Ds related to thermal hydraulics, such as flow induced vibration issues. These difficulties are facilitated by higher flow velocity in one loop. A scaled model water test of hot leg piping is preparing to obtain the prospect of solution. The integrated components may enhance the fretting wear of IHX tubes. The evaluation methods for the fretting wear in 12Cr steel and related experimental data are necessary to be investigated to ensure its integrity. A scaled model test is preparing to evaluate the vibration of IHX tubes induced by pump.

5.1.2. Helium gas-cooled fast reactor [30]

A carbon-dioxide gas-cooled FR and a helium gas-cooled FR were evaluated and compared with each other. For a gas-cooled FR, high temperature up to 850˚C with direct cycle system has a great potential to accomplish the development targets especially in economic competitiveness. Meanwhile the combination of the carbon-dioxide gas-cooled FR and gas turbine power generation would be very difficult to commercialize due to the problem of structural material corrosion. As the results, the helium gas-cooled FR with coated-particle fuel was selected because of its superiority of high plant efficiency.

Features of Helium-cooled Fast Reactor

① High-temperature/high thermal efficiency:	Multi-purpose utilization (Efficient power generation, Production of hydrogen etc.)
② Safety:	Potential for securing high degree of safety with passive safety features
③ Improvement of economic competitiveness:	Deletion of secondary cooling system and turbine buildings by adopting direct cycle system and common use of equipment by adoption of twin plant

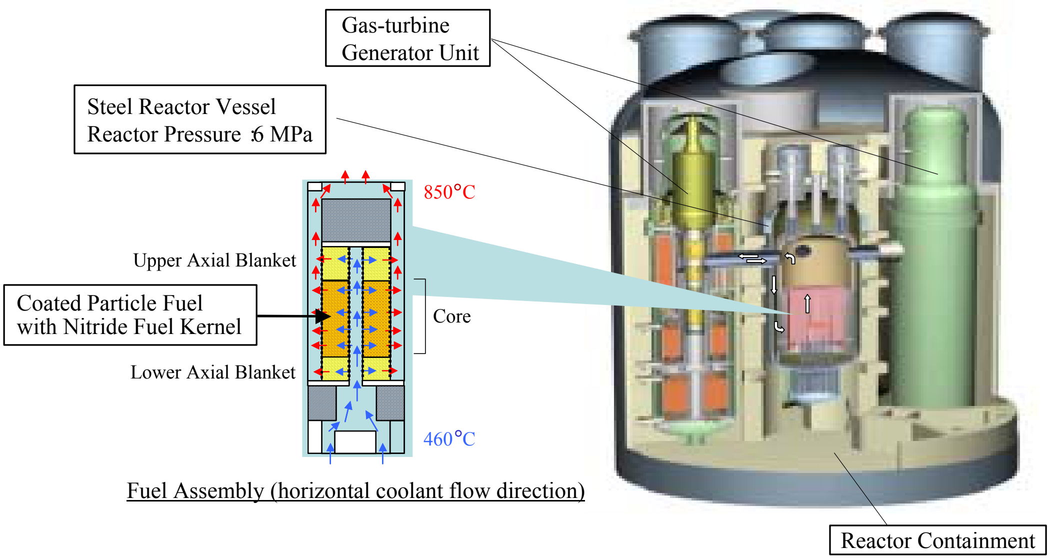

FIG. 11. Helium gas-cooled fast reactors.

With regard to the coated-particle fuel, a commercialized fuel concept for thermal neutron high-temperature gas reactors exists that uses spherical particles made by multi-layer coating an oxide fuel with high-density graphite layers or SiC to give the fuel particles high heat capacity and hard coating layer. However, as previous test data indicate that the coating layers will be damaged with a burn-up of less than 10,000 MWd/t in fast neutron environment. Thus TiN was selected as a candidate of most probable coating material in consideration of thermal, mechanical and irradiation characteristics against fast neutrons.

The most valuable feature of high temperature resisting coated-particle fuel is the high possibility of implementing countermeasures against reactor core disruption. Core melt and re-criticality should be avoided without any active component actuations even under depressurized accident conditions. Figure 11 shows the schematic of helium gas-cooled FR. The design study is in progress focusing on the achievement of a core melt proof reactor safety concept.

5.1.3. Lead-bismuth cooled fast reactor [31-40]

The heavy metal-cooled reactor is a relatively new plant concept that has only quite recently begun to be studied even in other advanced countries with the exception of its application in nuclear submarines in Russia. Therefore, FS evaluated heavy metal-cooled reactor concepts using data from previous studies conducted in Russia. At first, we selected a tank-type reactor because a loop-type reactor would require technically difficult measures to alleviate thermal stress for the piping due to high specific gravity. Next, we selected a medium-scale reactor because it would be difficult to achieve the economic competitiveness with a large-scale reactor due to the very large weight of core support structure and the difficulty of meeting the mandated standards to withstand earthquake.

With regard to the coolant, lead-bismuth was adopted because a lead-cooled reactor would require maintenance at high temperature of about 400˚C due to its high melting point. To compare the effect of coolant driving system on capital cost, the 55 MWe reactor of natural circulation type and the 75 MWe reactor of forced convection type were examined. The forced convection reactor system has slight economical advantage because the mass of unit power is smaller than that of the natural circulation type. Figure 12 depicts the forced convection type lead-bismuth-cooled FR.

For the lead-bismuth-cooled FR, the prevention of material corrosion in high temperature coolant is the most crucial issue of judging the technical feasibility. Therefore, tests have started to seek measures to control the corrosion of core and structural materials in collaboration with FZK and domestic universities. In the collaboration studies with FZK under stagnant lead-bismuth condition, ODS and 12Cr-steel showed good compatibility with lead-bismuth eutectic under a proper oxygen content of 10^{-6} mass% at 550˚C and bellow. However, the thickness of the oxide layer was getting thinner with temperature increase over 600˚C. Beyond 570˚C, dissolution attack was observed at some portions. It is estimated that the oxide layer becomes to loose its adhesion to the material. The application temperature range of the existing materials will be confirmed by corrosion tests. As the results obtained up to now, the development of a new anti-corrosive material is indispensable for the commercial use of the lead-bismuth-cooled FR.

Specific features;

- **Medium Tank Type & Modular**

  750MWe(1875MWt) x 2 module / unit

- **Reduction of the weight of the NSSS**
  - ✓ Without Secondary Loop (Two Circuit System)
  - ✓ Three-Dimensional Seismic Isolation
  - ✓ 12Cr-Steel with High Strength
  - ✓ Compact Steam Generator
- **Corrosive resistance of structural materials**
  - ✓ Selection of available materials or development of new steel types
  - ✓ Formation of protection coatings on steel surfaces;
  - ✓ Correction of impurities in coolant composition (especially, O_2 correction).

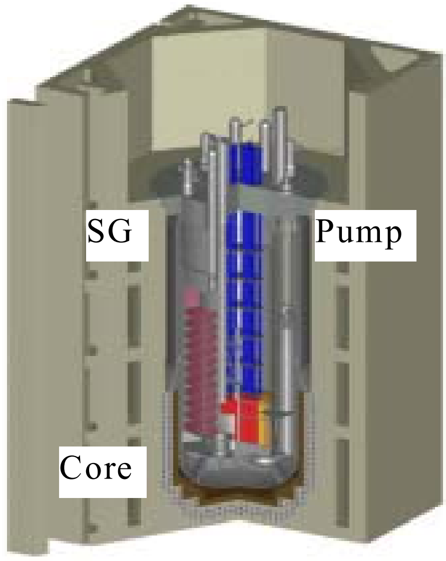

FIG. 12. Lead-bismuth-cooled fast reactor (medium-scale/tank-type/modular/forced convection.

The use of nitride fuel is necessary for the lead-bismuth-cooled FR to achieve high burn-up and breeding performance. With regard to nitride fuel, the important tasks to be addressed are: the collection of basic data through fuel irradiation tests, the economic concentration and collection of ^{15}N, and the analysis of high-temperature characteristics including the transitional characteristics such as the mechanism of nitrogen dissociation.

Since there are many issues to be solved for the lead-bismuth-cooled FR, it is considered that the long-term R&D works are needed until commercial use.

1.1.1. Water-cooled fast reactor [41-42]

The water-cooled fast reactor was designed by referring to a high conversion ratio BWR (Boiling Water Reactor) and increasing the density and void fraction of the core to increase the fuel volume ratio. In terms of economical competitiveness, it is considered that this concept will be equivalent or slightly superior to ABWR (Advanced BWR) because of the elimination of the recirculation pumps, the reduction in the containment vessel volume, etc.

The design base safety analysis and evaluation confirmed the possibility of attaining required safety level. However, it is necessary to clarify the core cooling performance during an accident and to evaluate the risk of re-criticality in the event of a core disruption, including the sequence of events and the behavior of the molten fuel during core disruption, and the performance of the containment function during re-criticality.

In terms of core fuel, Pu breeding ratio of about 1.04 can be achieved with low decontamination and high Pu enrichment fuel (about 35%). The multi-layer compound core system is adopted by using the fuel pin with inner blanket in the central core section. The Pu inventory of the compound system is about two times large compared to that of a sodium-cooled fast reactor.

It is necessary to confirm the thermal hydraulic and mechanical characteristics of dense core layouts as well as the safety performance during core disruption and to evaluate the possibility of further improvement in the burning capability of low-decontamination TRU fuel.

5.2. Fuel cycles

The current status of R&D for fuel cycle systems is shown in Figure 13. In order to accomplish the development targets, the fuel cycle system needs to be composed of simplified and rationalized processes. Furthermore, dealing MA (minor actinide) and LLFP as well as U and Pu in the closed cycle system, it is indispensable to adopt the innovative technologies to establish the fabrication process of low-decontamination factor TRU-bearing fuel, and to develop the remote operation and maintenance system in a hot cell facility. Main technologies for reprocessing have been broken down into categories such as U-recovery, U/Pu/Np-recovery and MA-recovery, and for fuel fabrication into categories such as fuel fabrication and stacking/compaction as shown in Figure 14 [43-46].

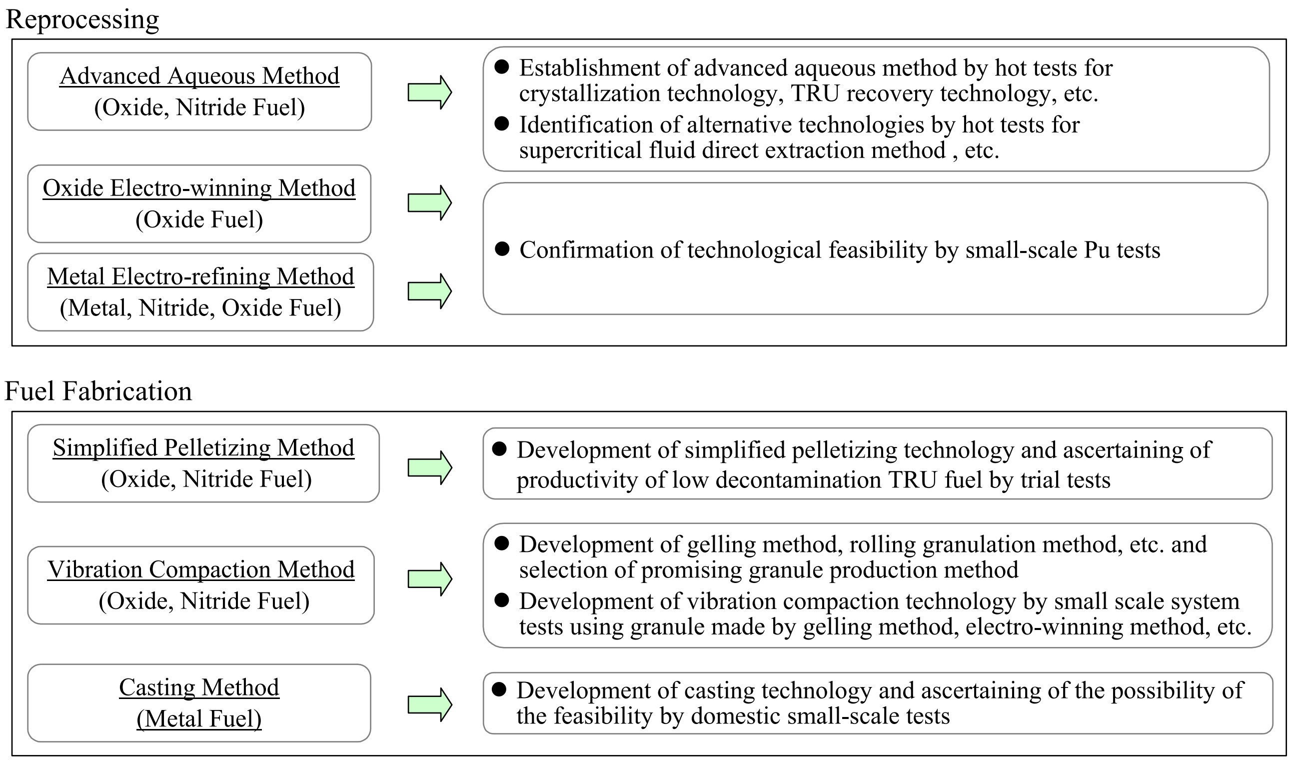

FIG. 13. Current status of R&D for fuel cycle systems.

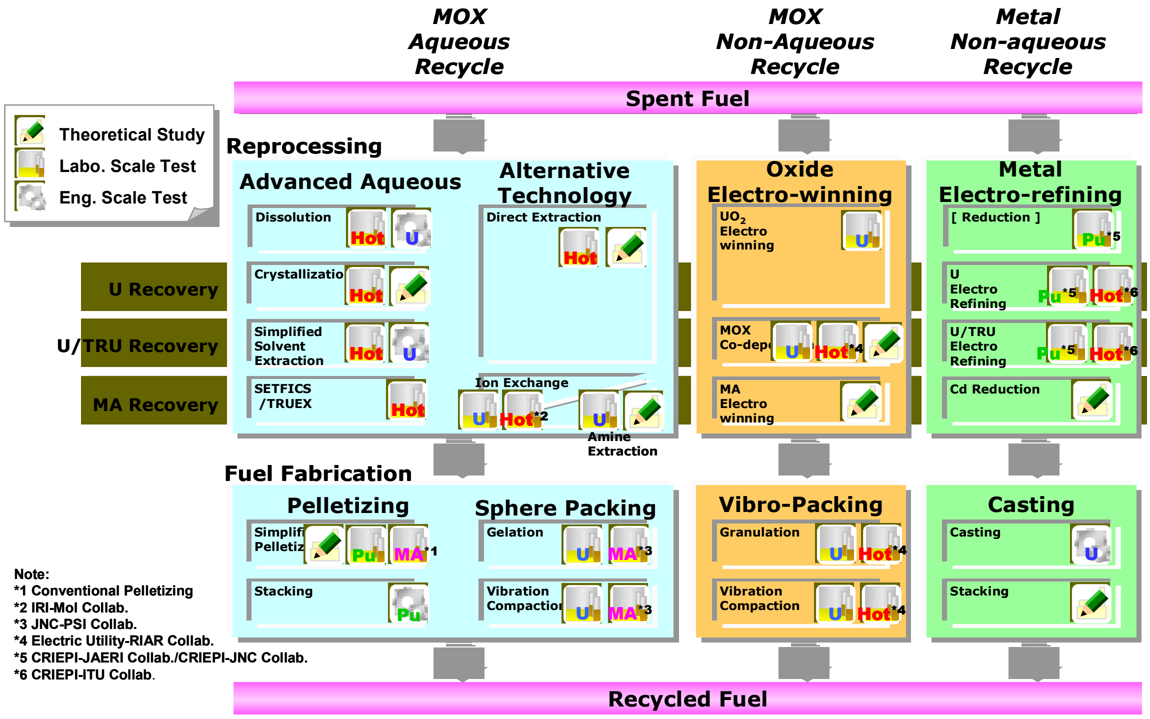

FIG. 14. Status of recycle technology.

5.2.1. Advanced aqueous process [47-55]

The advanced aqueous process consists of a simplified PUREX process with the addition of a uranium crystallization step and the SETFICS process as MA recovery process. The features of the process are the following:

— The purification process of U and Pu in the conventional process is eliminated, and
— U/Pu is co-extracted with Np with reasonable decontamination factors (DF) for recycle use,
— The uranium crystallization removes most of the bulk heavy metal at the head end and eliminates it from down stream processing,
— The main process stream is salt-free, which reduces the secondary waste,
— Neither separated Pu nor radiation-free nuclear material exists in any step of the entire fuel cycle.

In the simplified PUREX process, Np recovery in mixed U and Pu product solution has been demonstrated with DFs over 1000 in small-scale hot tests in Chemical Processing Facility (CPF). It should be studied to optimize U/Pu/Np recovery condition and the DF for fission products. In the crystallization process, the dissolved solution is cooled down and excess U is precipitated as a crystal of uranyl nitrate hexahydrate (UNH) according to the solubility at the low temperature. It is expected that the decontamination factors for fission products in the UNH product are approximately 100 from a simulated dissolver solution test. The combination of SETFICS/TRUEX process using TBP and CMPO is applied to the system as the MA recovery process. Small-scale hot tests were implemented to investigate the separation efficiency of MA from lanthanides. It was confirmed with cold tests that a salt-free reagent such as hydroxylamine nitrate (HAN) was applicable in this process.

Besides the above described process, some alternative techniques have been also investigated; supercritical fluid direct extraction method as the alternative for the dissolution, U recovery and U/TRU recovery, the amine extraction method as the alternative for the SETFICS, and the extraction chromatography method for the SETFICS.

5.2.2. Pyro-electrochemical process [56-58]

Candidates for advanced reprocessing are the modified pyro-electrochemical processes, which are based on Russian-RIAR and US-ANL methods. They offer several presumed advantages, the most important among them are the following:

— Ability to process refractory and hot fuels, due to the high solvating power of molten salts and the radiation-resistant features of the chemical reagents involved (no organic radio-sensitive molecules)
— Compactness (limited number of transformation steps – actinides is early recovered in the form suitable for recycling)

Some other features of the pyro-electrochemical process can be mentioned, like suitability for "Onsite processing".

5.2.2.1.Oxide electro-winning process

The reprocessing technology for oxide fuel is rather focused at the transition period from the current fuel cycle to the next advanced fuel cycle. The development of these key technologies focuses on the safe, reliable, industrial scale-up of electro-winning and refining systems including an extraction process for MA and LLFP.

Continuing efforts by RIAR have demonstrated a successful operation of the oxide electro-winning by using several kg of spent fuel from BOR-60. Japanese electric utilities and JNC are now trying to modify these processes. Especially, MOX co-deposition behavior has been demonstrated with highly-decontaminated UO_2 and PuO_2. However, it is known that some of fission products and cladding material disturb the MOX co-deposition behavior.

5.2.2.2.Metal electro-refining process

The R&D on metal electro-refining process has been principally designated to CRIEPI. Metal electro-refining is more effective for fast reactor metallic fuel cycle. Applying metal electro-refining process to the fast reactor oxide base fuel cycle, additional processes for the initial reduction of MOX and the final oxidation of metal are required. Fundamental studies are still required to adjust the electro-winning condition.

Development of long-life component materials, including crucible material, is an issue for realization of the dry process because of the corrosive and high temperature operating conditions. Safeguard ability of the dry process should be assured with the real time monitoring equipment and inspection system. The loss of fissile materials to waste should be minimized and the recovery of MA can be further optimized. The treatment of chloric type wastes has to be guaranteed for long-term stable storage. Concept optimization for industrial-scale spent fuel pyroprocessing is important to reveal weakness of existing pyroprocesses and clear the direction of the improvement.

JNC is arranging testing infrastructures in Tokai works for metal electro-refining as well as oxide electro-winning. Collaboration programs with domestic and overseas partners are in progress. The glove box equipment is being prepared by CRIEPI at CPF in JNC-Tokai Works to make integral Pu and U experiments related to metal electro-refining.

In small-scale process feasibility tests, which have been carried out by CRIEPI in collaboration with the Institute for Transuranium Elements (ITU) of the EU, the electro-refining tests for U-Pu-Zr ternary alloy fuel containing MA (non-irradiated) were performed and fundamental data, such as recovery ratio, were obtained. Also, tests for recovery of Pu by liquid Cd cathode were carried out using U/Pu ratio in salt as a parameter. It was confirmed that recovery of heavy metals at a concentration exceeding the design value (10wt%) is possible and the possibility of rationalization of the system was confirmed. Moreover, it should be studied to optimize U/Pu/MA recovery condition under fission product in liquid Cd cathode.

5.2.3. Fuel fabrication [59-60]

Fuel fabrication for the advanced cycle must be as simple as possible and suitable for the massive remote operation to handle radioactive materials, which are recovered from the reprocessing with low decontamination factors. Three candidates for fuel fabrication process are being investigated in FS; a simplified pellet process, vibro-packed process using particulate fuel and metal casting process. R&D for metal casing process is in progress by CRIEPI to modify the casting fuel fabrication method.

5.2.3.1.Simplified pellet process

MOX pellet fabrication technology based on glove box confinement has been verified in highly decontaminated plutonium recycling system in the commercial LWR and fast reactor fuel cycle. However, this process must be modified to fabricate low DF fuels in a remote operation mode. Simplified pellet process is the shortest route in which MOX powder adjusted Pu content is co-converted directly by microwave heating process from Pu, U and Np nitric acid solutions for the next pelletizing process with minimum pre-treatment of powder. The key in this technology is to prepare the well-homogenized and controlled powder to obtain a high throughput.

When the simplified pellet process is applied to low DF products, the pellet design specification should be relaxed to realize the operability of the fabrication system in a hot cell facility and to match an increase of impurity level. In addition, it is necessary that fabricability of MA-bearing pellet fuel is established under low DF condition. JNC is conducting basic parameter tests to optimize powder characteristics and fabricate MOX pellet with the preliminarily simplified process.

5.2.3.2.Sphere and vibro-packed fuel fabrication

Sphere and vibro-packed fuel fabrication method is a most important technology common to the particulate fuel products from both aqueous and dry reprocessing in advanced fuel cycle. The concept of vibro-packed fuel itself was introduced about 40 years ago. The key issue is the selection and optimization of particulate MOX-MA fuel fabrication methods. These are gel precipitation, dry granulation process in simplified PUREX and MOX co-precipitation in oxide electro-winning.

The controlled fabrication method of two or three size distribution of granular particles is important to achieve an equivalent smear density with MOX pellet pin of about 80%TD. The challengeable technology is to get the smaller size particle of higher Pu content MOX-MA fuel controlled less than 100μm in diameter by remote operation. The experiences in BNFL and RIAR for the vibro-packed fuel suggest that fuel pin quality assurance should be optimized for the continuous operation of manufacturing and inspection process. Laser scanning system to guarantee the quality of particle fuel and three-dimensional CT scan system to check the smear density distribution in a fuel pin are still under basic investigation. The prevention of fuel-cladding chemical interaction (FCCI) is another issue. Controlling oxygen potential under an irradiation condition is necessary to achieve high burn-up capability. Both the initial conditioning of oxygen-to-metal ratio of MOX particulate and the mixing of oxygen getter with fuel particle are candidates.

5.2.4. Fuel cycle options

Considering the above elementary technology results and the experience of conventional reprocessing development, three fuel cycle processing technology options have been studied.

5.2.4.1.Advanced aqueous + Simplified pelletizing or vibration compaction

The conceptual design study on the system with plant capacity of 200t/y has been confirmed to satisfy the development targets. Moreover, regarding a small-scale plant of about 50t/y, in order to improve economic competitiveness, the following tests are being conducted on a laboratory-scale using spent fuel:

— Technology for alternating key parts of the system
— Technology for fabrication of MA-containing MOX pellets by remote operation

R&D is necessary concerning further reduction of waste, simplification of the system, and recovery of LLFP elements.

5.2.4.2.Oxide electro-winning + Vibration compaction

The conceptual design study with plant capacity of 50t/y has been confirmed. The following R&D is essential in order to confirm the possibility of technical feasibility of the system:

— Confirmation of the technical feasibility of the key technology (MOX co-deposition, MA recovery, etc.)
— Technology for manufacturing of uranium-added fine metal particles that control the performance of vibration compacted fuel

R&D is necessary concerning salt waste disposal and recovery technology for LLFP elements.

5.2.4.3.Metal electro-refining + Casting

The conceptual design study with plant capacity of 50t/y has been confirmed to satisfy the development targets. In order to improve performance of the key technology of the system, the reaction speed and recovery rate tests are being conducted on a laboratory-scale using the fresh MOX fuel. R&D is necessary concerning salt waste disposal and recovery technology for LLFP elements.

6. CONCLUSION

FS has been in progress aiming at commercialization of promising FR cycle system for base road power supply in Japan. Reactor core and plant system designs with various kinds of fuels and coolants have been studied to clarify the potential performance and the technical feasibility of FR cycle concepts. Crucial R&D items have been found out for each FR cycle concept and several experimental and/or analytical works have been underway. The perspective for the promising candidate concepts will be clarified preliminary at the end of March 2004, based on the on-going design studies and R&D results. The comprehensive evaluation method is expected to offer transparent and objective evidences to support the estimation. The significance on the international collaboration becomes greater than so far.

7. REFERENCES

[1] AIZAWA, K., "R&D Activities on FR Cycle Technologies for Transmutation of TRU and LLFP by JNC," Intl Seminar on Advanced Nuclear Energy Systems toward Zero Release of Radioactive Waste, Susono, Japan (Nov. 2000).

[2] NODA, H., et al., "Feasibility Study on Commercialized FR Cycle Systems in Japan - the Results in the First Phase and Future Plans of the Study," Global 2001, Paris, France (Sep. 2001).

[3] AIZAWA, K., "Prospective View from Feasibility Study on Commercialized FR Cycle System," The 35th JAIF Annual Conf., Saitama, Japan (Apr. 2002).

[4] NODA, H., "The Feasibility Study on Commercialized Fast Reactor Cycle System," the 17th KAIF/KNS Annual Conference, Korea (Apr. 2002).

[5] NODA, H., "Current Status of the Feasibility Study on Commercialized Fast Reactor Cycle System," NTHAS3: Third Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety, Kyeongju, Korea (Oct. 2002).

[6] ICHIMIYA, M., "Design Study on Advanced Fast Reactor Cycle System in Japan," the 18th KAIF/KNS Annual Conf., Korea (Apr. 2003).

[7] HIRAI, K., "Japanese Utilities' Activity on the Development for Future Reactors," NTHAS3: Third Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety, Kyeongju, Korea (Oct. 2002).

[8] AIZAWA, K., "R&D for Fast Reactor Fuel Cycle Technologies in Japan," Global 2001, Paris, France (Sep. 2001).

[9] FUJII, S., et al., "Evaluation Methodology and Prospective Introduction Scenarios of FR Cycle Systems," to be published in GENES4/ANP2003, (Proc. Kyoto, 2003).

[10] SHINODA, Y., et al., "Development of Characteristic Evaluation Method on FR Cycle System," Intl Congress on Advances in Nuclear Power Plants (ICAPP) Hollywood, Florida, USA (2002).

[11] KOTAKE, S., et al., "The R&D Issues Necessary to achieve the Safety Design of Commercialized Liquid Metal Cooled Fast Reactors," OECD/NEA/CSNI Workshop on Advanced Nuclear Rector Safety Issues and Research Needs, Paris, France (2002).

[12] NIBE, N., et al., "Feasibility Studies on Commercialized Fast Breeder Reactor System (1) -Sodium Cooled Fast Breeder Reactor-," Proc. of SMiRT 16, Washington, DC, USA (2001).

[13] SHIMAKAWA, Y., et al., "An Innovative Concept of Sodium-Cooled Reactor Pursuing High Economic Competitiveness," Nuclear Technology, Vol. 140, No.1 (2002).

[14] MIZUNO, T., et al., "Advanced Metal Fuel Core Design Study of Sodium Cooled Reactors in Current Study on Commercialized Fast Reactor Cycle Systems in Japan," Intl Congress on Advances in Nuclear Power Plants (ICAPP), Cordoba, Spain (May 2003).

[15] OHKI, S. "Comparative Study for Minor Actinide Transmutation in Various Fast Reactor Core Concepts," Proc. Sixth Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, Madrid, Spain (Dec. 2000).
[16] [16] Takagi, N., et al., "Feasibility and Challenges of LLFP Transmutation in Fast Reactor," Global 2001, Paris, France (Sep. 2001).
[17] OHKI, S., et al., "Transmutation of Cesium-135 with Fast Reactors," Proc. Seventh Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, Jeju, Korea, (Oct. 2002).
[18] OHKI, S., et al., "Analysis of Cobalt-60 Production Experiment in the Fast Reactor PHENIX," Proc. Int. Conf. PHYSOR 2002, Seoul, Korea (Oct. 2002).
[19] UKAI, S., et al., "Development of 9Cr-ODS martensitic steel claddings for fuel pins by means of ferrite to austenite phase transformation," J. Nucl. Scie. and Technol., Vol.39, No.7, pp.778-788 (2002).
[20] UKAI, S., et al., "Characterization of high temperature creep properties in recrystallized 12Cr-ODS ferritic steel claddings," J. Nucl. Scie. and Technol., Vol.39, No.8, pp.872-879 (2002).
[21] KITAMURA, S., MORISHITA, M., "Design Method of Vertical Component Isolation System," SMiRT-16 (2001).
[22] KITAMURA, S., MORISHITA, M., "Design Method of Vertical Component Isolation System," ASME PVP-Vol. 445-2, pp55-60 (2002).
[23] MORISHITA. M., KITAMURA, S., et al., "Structure of 3-Dimensinal Isolated FBR Plamt with vertical Component Isolation System," SMiRT-17 (2003).
[24] KITAMURA S., Somaki T., et al., "Experimental Study on Coned Disk Springs for Vertical Seismic Isolation System," SMiRT-17 (2003).
[25] KISOHARA, N., et al., "Innovative Sodium/Water Reaction Mitigation System for a large-Sized Steam Generator of Fast Breeder Reactor," Intl Congress on Advances in Nuclear Power Plants (ICAPP), Cordoba, Spain (May 2003).
[26] MURAMATSU, T., et al., "Numerical Investigations on In-Vessel Thermohydraulic Characteristics Related to Gas Entrainment Phenomena of Sodium-Cooled Fast Reactors," Intl Congress on Advances in Nuclear Power Plants (ICAPP), Cordoba, Spain (2003).
[27] KIMURA, N., et al., "Experimental Study on Flow Optimization in Upper Plenum of Reactor Vessel for a Compact Sodium Cooled Fast Reactor," to be published in Proc. of the 10th Intl Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea (Oct. 2003).
[28] ITO, K., et al., "Numerical Simulation of Free Surface Vortex in Cylindrical Tank," to be published in Proc. of 10th Int. Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea (Oct. 2003).
[29] NIWA, H., et al., "LMFBR Design and its Evolution: (3) Safety System Design of LMFBR," to be published in GENES4/ANP2003 (Proc. Kyoto, 2003).
[30] KISO, Y., et al., "Feasibility Study on Commercialized Fast Breeder Reactor System (2) -Gas-Cooled High Temperature FBR-," Proc. of SMiRT 16, Washington, DC, USA (2001).
[31] ENUMA., Y., et al., "HLMC Fast Reactor with Complete Natural Circulation," Proc. of 10th Intl Conf. on Nuclear Engineering (ICONE-10), Washington, DC, USA (Apr. 2002).
[32] MIHARA, T., et al., "Feasibility Study on Commercialized Fast Breeder Reactor System (3) -HLMC Fast Reactor-," Proc. of SMiRT 16, Washington, DC, USA (2001).
[33] ENUMA, Y., "Conceptual Design of a Medium Scale Lead-Bismuth Cooled Fast Reactor," Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).

[34] CHIKAZAWA, Y., et al., “Conceptual Design of a Small Lead-Bismuth Cooled Reactor,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[35] AOTO, K., et al., “Compatibility of Japanese FBR steels in High Temperature LBE,” Proc. ANS2002 Winter Meeting (2002).
[36] SAKAI, T., et al., “System Analysis for Decay Heat Removal in Lead-Bismuth Cooled Natural Circulated Reactors,” Proc of Intl Congress on Advanced Nuclear Power Plants (ICAPP), Florida, USA (Jun. 2002).
[37] ITO, K., et al., “Oxygen Diffusion Analysis of Lead-Bismuth-Cooled Natural-Circulation Reactor,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[38] SAKAI, T., et al., “System Analysis for Lead-Bismuth-Cooled Natural Circulation Reactors,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[39] FURUKAWA, T. et al., “Corrosion Properties of Japanese FBR Materials in Stagnant Pb-Bi at Elevated Temperature,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[40] LOEWEN, E., et al., “Investigation of Polonium Removal Systems for Lead-Bismuth Cooled Fast Reactors Using a Tellurium Surrogate,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[41] OKUBO, T., et al., “Advanced Concepts of Reduced-Moderation Water Reactor (RMWR) for Plutonium Multiple Recycling,” Global 2001, Paris, France (Sep. 2001).
[42] OKA, Y., et al., “Design Concept of One-Through Cycle Supercritical-Pressure Light Water Cooled Reactors,” The First Intl Symposium on Supercritical Water-cooled Reactors, Design and Technology (SRC-2000), Tokyo, Japan (Nov. 2000).
[43] FUNASAKA, H., et al., “Present Status and Prospects in the FR Fuel Cycle System in Japan,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[44] NOMURA, S., et al., “Development of Challengeable Reprocessing and Fuel Fabrication Technologies for Advanced Fast Reactor Fuel Cycle,” Global 2001, Paris, France (Sep. 2001).
[45] TAKAKI, N., “ORIENT-CYCLE -Evolutional Recycle Concepts with Fast Reactor for Minimizing Hign-Level Waste-,“ OECD/NEA Seventh Information Exchange Meeting on Actinide and Fission Product Partitioning & Transmutation, Jeju, Republic of Korea, (Oct. 2002).
[46] NAGAI, T., “The Excellent Fuel Cycle Technology in Nuclear Proliferation Resistance,” Proc. of 10th Intl Conf. on Nuclear Engineering (ICONE-10), Washington, DC, USA (Apr. 2002).
[47] TANAKA, H., et al., “Design Study on Advanced Reprocessing Systems for FR Fuel Cycle,” Global 2001, Paris, France (Sep. 2001).
[48] TAKATA, T., “Conceptual Design Study on Advanced Aqueous Reprocessing System for Fast Reactor Fuel Cycle,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[49] KURISAKA, K., et al., “Risk Analysis of the Aqueous Fast Reactor Fuel Cycle Facility in the Conceptual Design Stage,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[50] AOSHIMA, A., “Renovation of Chemical Processing Facility for Development of Advanced Fast Reactor Fuel Cycle System in JNC,” Proc. of 10th Intl Conf. on Nuclear Engineering (ICONE-10), Washington, DC, USA (Apr. 2002).

[51] HAYASHI, N., et al., “Development of the MAREC Process for HLLW Partitioning Using a Novel Silica-Based CMPO Extraction Resin,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[52] MORIMOTO, K., et al., “Influence of Np on Sintering Behavior and Phase Separation for $(Pu,Np,U)O_{2-x}$,” Global 2001, Paris, France (Sep. 2001).
[53] KOMA, Y., et al., “Separation Process of Long-Lived Radionuclides for Advanced Fuel Recycling,” Global 2001, Paris, France (Sep. 2001).
[54] KOMA, Y., et al., “Possible Routes for Recovery of Some Fission Products from PUREX Process,” Proc. of the RRTD 2nd International Workshop on Nuclear Fuel Cycle –Nuclear Fuel Cycle from the Viewpoint of Disposal SIte Utilization-, Aomori, Japan (Mar. 2003).
[55] FUJII, K., et al., “Effects of Separation of Minor Actinide, Cesium and Strontium on High-level Radioactive Waste Disposal,” Proc. of the RRTD 2nd International Workshop on Nuclear Fuel Cycle –Nuclear Fuel Cycle from the Viewpoint of Disposal SIte Utilization-, Aomori, Japan (Mar. 2003).
[56] OKAMURA, N., et al., “Proposal of Materials Transfer Capability Evaluation Model and its Application to Pyrochemical Reprocessing Plant,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[57] YONEZAWA, S., “Computer Simulation of Transport and Maintenance Methods in a Pyrochemical Reprocessing Plant Design,” Proc. of 11th Intl Conf. on Nuclear Engineering (ICONE-11), Tokyo, Japan (Apr. 2003).
[58] HEBDITC, et al., “Concept Optimization for Industrial-Scale Spent Fuel Pyroprocessing,” Global2001, Paris, France (Sep. 2001).
[59] TANAKA, K., et al., “Conceptual Design Study of Advanced Fuel Fabrication Systems,” Global2001, Paris, France (Sep. 2001).
[60] MIYAMOTO, H., et al., “Development of Compaction Technology of Vibro-compaction Fuel Fabrication - Investigation of Compaction Behavior and Vibration Parameter Effect on Sphere Fuel,” Global 2001, Paris, France (Sep. 2001).

NUCLEAR POWER DEVELOPMENT ON THE BASIS OF NEW NUCLEAR REACTOR AND FUEL CYCLE CONCEPTS

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Abstract. Much of the global demand for electricity and some demand for heat can be met by a nuclear technology that will comply with the safety, environmental and economic requirements of a large power industry. Nuclear power can grow on a large scale based primarily on big nuclear plants with fast reactors. The key requirements among those placed on the reactor and fuel cycle technologies include: efficient utilisation of accumulated Pu and reduction of specific U consumption by an order of magnitude or more; natural safety – deterministic exclusion of accidents involving large radioactive releases, balance between the radiation hazards of radioactive waste subject to burial and of uranium extracted from the earth; resistance to proliferation of nuclear weapons; reduction in the cost of new plants relative to modern LWRs. This presentation describes the work done on designing a plant with a demonstration lead-cooled 300 MWe reactor (BREST-OD-300) and on experimental validation of the adopted reactor and fuel cycle design.

1. INTRODUCTION

The "*Strategy of nuclear power development in Russia in the first half of the 21st century*" [1] maintains that the experience in using nuclear energy gained by mankind over 50 years provides a sound basis for developing and demonstrating early in this century a technology of fast reactors with a closed nuclear fuel cycle, that will meet today's requirements of large-scale energy production in terms of economic, environmental and safety characteristics, and will be capable of attaining the goals identified by nuclear scientists back in the 1940s.

Studies carried out by Russian scientists and engineers in the last 15 years [2] laid a groundwork for the Initiative "On energy supply for sustained development of mankind, radical solution of nuclear weapons proliferation problems, and global environmental improvement", which was voiced by President Putin at the UN Millennium Summit on September 6, 2000, as an invitation to international cooperation in these vitally important areas. With this aim in view, Minatom initiated an international project at the IAEA, referred to as INPRO.

The lessons drawn from the fifty-year experience and consideration of the new conditions suggest that the following main requirements should be placed on the NPP with a fast reactor and its closed fuel cycle:

— equilibrium fuel composition (BR=CBR≈1); provision of a Th blanket to produce U for thermal reactors at some later date, when cheap ^{238}U is exhausted;
— strong and convincing safety case for long-term and large-scale production of nuclear electricity, with accidents involving catastrophic radioactive releases deterministically excluded despite all possible human errors, equipment failures and external impacts;
— a strong and convincing safety case for burying radwaste for many thousands of years without upsetting the natural radiation balance;

— elimination of the weapons-grade material generation channels in today's nuclear power, such as radiochemical separation of Pu and, at a later point, isotope enrichment of U, - to go hand in hand with improvements in the international nonproliferation regime and physical protection measures;
— lower plant cost relative to the existing power units so as to make nuclear competitive with conventional energy sources.

Fast reactors with U-Pu fuel have a unique excess of neutrons, which is their fundamental physical resource for meeting the key requirements. Moreover, it allows finding an adequate technical concept (fuel, coolant, design, etc.) for the fast reactor, which will not be too different from the existing civil and military technologies but will be consistently based on the principles of natural safety, which means essentially reliance on the laws of nature.

To prove the feasibility of making an NPP consistent with the above requirements of a large-scale power industry, it is planned to build an experimental plant at the Beloyarsk NPP site, which will have a demonstration 300 MWe lead-cooled fast reactor (BREST-OD-300) with an on-site nuclear fuel cycle and a radwaste treatment complex.

The main working objectives of the BREST-OD-300 plant and its on-site fuel cycle include:

— exclusion of prompt criticality excursion by virtue of core physics and design (equilibrium fuel composition and reactivity margin of ~ß);
— demonstration of reactor resistance to severe accidents with and without operation of the control and protection system:
  - input of the whole reactivity margin;
  - trip of primary and secondary pumps;
  - rupture of steam generator tubes;
  - freezing - unfreezing;
  - coincidence of accidents;
  - extreme accidents;
  - all kinds of “made-up” accidents;
— launching of the on-site closed fuel cycle;
— operational refinement of the radwaste treatment processes.

In 2002, the BREST-OD-300 design was developed for the Beloyarsk NPP site, including [3]:

— engineering design of the BREST–OD–300 reactor facility:
  - reactor facility;
  - steam generator;
  - pump;
  - upper plate;
  - reactor vault;
  - reloading machine.
— engineering design of reactor facility systems for:
  - heating;
  - coolant intake, conditioning and injection;
  - pressure compensation;
  - radioactive gas treatment;
  - coolant treatment with gas mixtures;
  - air cooling of the vault;
  - normal and emergency cooling;

- confinement of steam generator leaks;

— NPP design:
  - general layout;
  - process design features;
  - main building;
  - turbine hall and secondary circuit;
  - structural design features;
  - construction management plan;
  - preliminary safety case;
  - environmental qualification;
  - survey for construction.

Engineering design of fuel cycle equipment has been completed for:

— cutting of fuel assemblies;
— fuel regeneration;
— fabrication of fuel rods;
— fabrication of fuel assemblies.

Engineering design of radwaste treatment facilities is finished.

The general view of the NPP with its on-site fuel cycle and the design of the BREST-OD-300 reactor are given in Figures 1 and 2. The reactor characteristics are found in Table I.

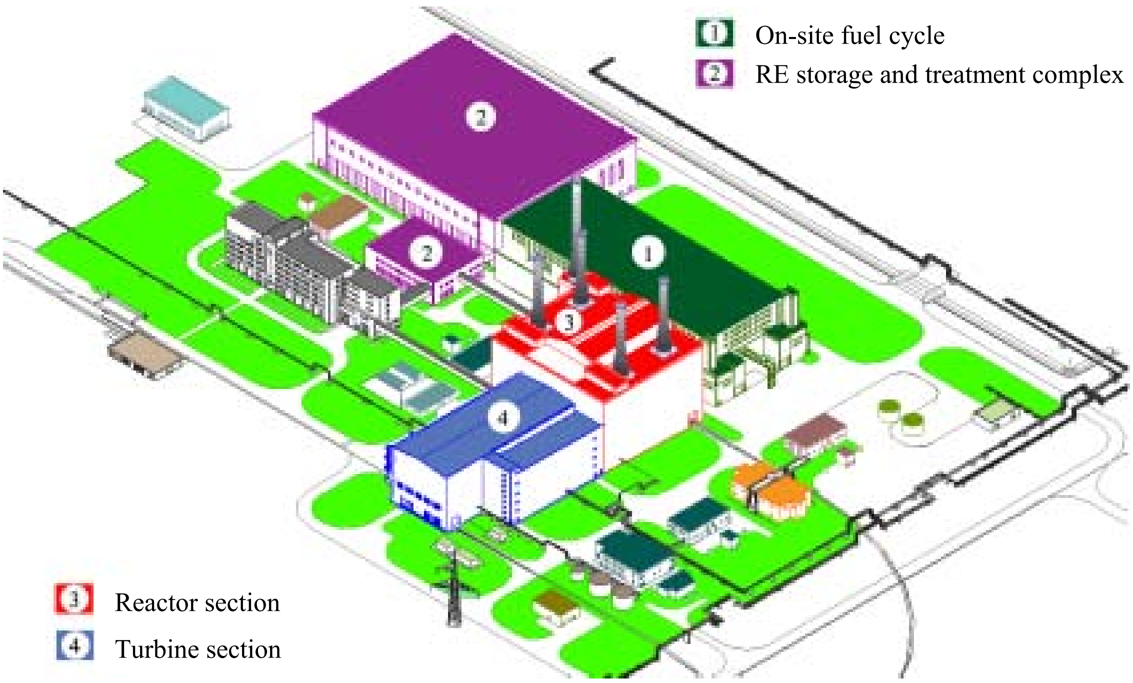
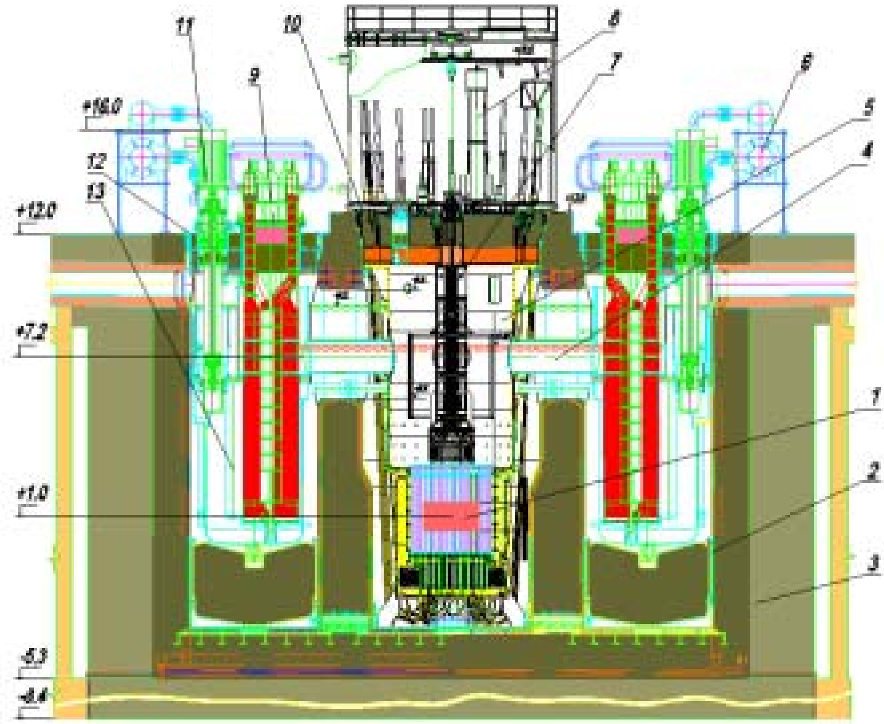


FIG. 1. General view of the NPP.



1 - core	*5 - core barrel*	*9 - steam generator*	*13 - filter*
2 - vessel	*6 - cooling system*	*10 - upper plate*	
3 - reactor vault	*7 - instrumentation column*	*11 - main circulation pump*	
4 - header	*8 - in-pile reloading machine*	*12 - SG-MCP unit*	

FIG. 2. BREST-OD-300 Reactor.

2. PRINCIPAL SAFETY FEATURES

To provide stability of BREST-OD-300 reactor in off-normal conditions, the core and the cooling circuits have been designed as follows [4]:

— fuel is mixed uranium-plutonium mononitride (UN + PuN) which features high density $\gamma \geq 13$ g/cm^3, heat conductivity $\lambda \sim 18$ W/(m·K), melting temperature T_{melt}=2800°C and phase transition temperature T_{phase}=1300°C;

— coolant is liquid lead which does not enter into exothermic reaction with water, air and structural materials, does not catch fire, is resistant to radiation, is low-activated, and allows heat removal at low pressure and with a large boiling margin ($T_{boil} \sim 2000$°C at P~1 MPa);

— lead bond between fuel and cladding excludes their thermal-mechanical interaction, provides high heat conductivity of fuel rods and low operating temperature of fuel ($T_{av} \sim 620$°C and $T_{max} < 900$°C), and reduces release of fission gas and its pressure on the cladding as burning progresses;

— core composition and geometry and fuel properties provide for full reproduction of plutonium in the core (BR=CBR ~1), ensure small reactivity variation with fuel burnup ($\Delta\rho_{burn} << \beta_{eff}$), small power and total reactivity effects ($\Delta\rho_{tot} \sim \beta_{eff}$);

Table I. Technical characteristics of BREST-OD-300 and BREST-1200 reactors

Characteristic	BREST-OD-300	BREST-1200
Thermal power, MW	700	2800
Electric power, MW	300	1200
Core fuel	UN+PuN	UN+PuN
Fuel inventory, (U+Pu+MA)N, t	17.6	68
Fuel lifetime, eff. Days	1500	1500÷1800
Refuelling interval, eff. days	300	300
CBR	~1.05	~1.05
Average in/out Pb temperature, °C	420/540	420/540
Maximum coolant velocity, m/s	1.7	1.7
$T_{clad}^{\max}$ with allowance for overheating, °C	650	650
$T_{fuel}^{\max}$ with allowance for overheating, °C	870	880
Steam temperature at the steam generator outlet, C°	525	520
Number of steam generators/pumps	4/4	8/8
Superheated steam pressure at the SG outlet, MPa	25	24.5
Design service life, years	30	60

— a "sparse" square fuel lattice and shroudless fuel assemblies provide a large flow area for the coolant and high natural circulation, and help avoid loss of heat removal in the event of local flow blockage at the FA inlet;

— three-zone fuel profiling in the core – which is provided by the use of fuel rods with different diameters but the same fuel composition and rod pitch in FA (rods with a smaller diameter placed in the centre of the core and larger ones, at its periphery) - ensures uniform lead temperature gain and cladding temperature distribution across the core, allows stabilisation of these parameters and increases temperature margins;

— lead reflector, instead of uranium blanket, flattens power distribution, provides large negative reactivity effect in case of low lead level in the reactor, reduces density coefficient of reactivity, and rules out production of weapons-grade Pu;

— passive coolant flow feedback to reactivity is implemented by lead-filled channels, with levels therein depending on the lead head at the core inlet;

— passive threshold feedback to reactivity from coolant flow and temperature is implemented by hydraulic control elements which provide passive reactor shutdown in case of loss of circulation or high coolant temperature at the core outlet;

— lead circuit is designed to have a great heat storage capacity and high flow inertia, and is provided with a flow bypass to ensure natural circulation of lead in case of pump trips;
— passive removal of decay heat directly from the lead circuit by a system with natural air circulation for an unlimited time;
— configuration and parameters of the steam-water circuit prevent lead freezing;
— large volume of the gas plenum, steam discharge from the gas plenum to emergency condensers and on, to the stack, and a hydraulic valve installed on the reactor lid, prevent circuit overpressure and lid tear-off in a very unlikely event of multiple rupture of SG tubes.

The discussion below addresses accidents caused by initiating events that none of the existing reactors, including those under development, can survive (i.e. beyond-design-basis accidents), and accidents specific to BREST:

— input of maximum operating reactivity (TOP WS),
— total loss of forced circulation of lead (LOF WS),
— total loss of heat sink to the secondary circuit (LHS WS),
— overcooling of the primary circuit (OVC WS),
— coincident occurrence of initiating events (WS),
— loss of steam generator integrity,
— loss of integrity of the lead circuit and reactor building.

Several temperature levels indicative of the extent of fuel damage were chosen for accident analysis:

— fuel cladding temperature up to T_{clad}=800°C and fuel pellet temperature up to T_{pel}=1300°C - the core remains serviceable due to low stresses in fuel claddings;
— fuel cladding temperature T_{clad} ≤900 ^{o}C (for a short time, ~20s) – fuel rods remain intact; cladding temperature in the range from 900 to 1200 ^{o}C - fuel cladding failure, release of gaseous and volatile fission products into the circuit, damage to some structural elements of the reactor (an accident of so-called “economic” class, with loss of reactor core or reactor as a whole);
— T_{clad}>1200°C - damage to the reactor core, melting of fuel claddings (T_{melt}~1500°C), lead boiling (T_{boil}~2000°C), decomposition (T_{ph}>1600°C) and melting (T_{melt}~2800°C) of mononitride fuel, damage to many reactor structures and possible loss of primary circuit integrity with the ensuing release of radioactivity.

All these initiating events and reactor response to them are discussed in detail in Ref. [4]. Let us turn to the accident involving failure of external safety barriers.

One of the reasons behind the choice of heavy coolant rather than sodium for fast reactors is associated with accidents involving failure of external safety barriers (reactor lid, building), leading to long-term exposure of coolant to the atmosphere. Deterministic demonstration of the plant safety requires analysis of even such unlikely, but possible, events, proving that the extremely severe consequences of these accidents are ruled out.

No detailed analysis has been performed so far for such accidents at fast reactors. Despite the scarcity of the input data required for such analysis, rough calculations were performed to assess the consequences of the accident according to one of its possible scenarios for the 300 MW(e) fast reactors with sodium and lead coolants.

The essential differences in these two cases stem from the self-sustaining burning of Na when in contact with air. This event will cause substantial heat release equal to or exceeding residual heat release, high pressure of Na vapours (3 mm Hg at 500 ^{o}C as compared to 10^{-2} mm Hg with Pb), high radioactivity of sodium proper, and loss of fuel cooling as a result of Na burning.

As postulated, the accident involves reactor lid tear-off, destruction of the building, lead temperature increase to 700 ^{0}C, and sodium temperature rise to 500 ^{0}C. It is assumed that temperatures remain at these levels for four days, resulting in the maximum radioactivity release in case of lead coolant and minimum release in case of sodium, because lead temperature will decrease while sodium temperature will rise when heat removal can only rely on radiation and air convection. Calculations and experiments showed that when in contact with air, lead does not catch fire under T=1200 ^{0}C. Therefore, fuel rods are assumed to remain intact and radioactive release from fuel does not exceed the design limit.

It was assumed in both cases that chain reaction was suppressed, but the personnel failed to terminate radioactive release. The release from gas plenum in the beginning of the accident was disregarded in the analysis.

Lead entrainment was assessed from experimental data obtained in the studies on liquid lead evaporation and oxidation in the first 10 hours of lead-air contact when the rate of evaporation is at its maximum. In the accident under consideration, the rate of lead entrainment was 22 g/s. Under the worst weather conditions, lead concentration near the ground at a distance of 10 km from the site would be 0.02 mg/m^3, which is equivalent to 2 MPC (maximum permissible concentration) in terms of the chemical toxicity of lead (lead MPC = 0.01 mg/m^3).

Estimation of maximum release of hazardous radioactive nuclides in the course of the accident (during tens of days), made under the assumption that lead of Grade C00 will be used in BREST and that there will be no in-service purification of lead to remove these radionuclides, showed that total release of radioactivity would amount to ~2300 Ci. It should be noted that potential radiation hazard of the daily release of polonium - a mere 3 Ci - is equivalent to the radiation hazard of the cumulative release of all other radionuclides during the accident.

Release of Po from lead may depend on formation of the intermetallic compound Po-Pb, occurrence of oxide films on the lead surface and on other factors that are yet to be investigated. They were not considered in this analysis.

The estimated release of radioactivity in the postulated accident at BREST reactor corresponds roughly to Level 5 according to the international scale of events at nuclear plants (i.e. an accident involving risk to environment). Coolant purification to remove Po and other radionuclides would allow bringing the accident consequences down to Level 4 or even Level 3. However, if this accident develops to cause fuel damage, radioactive release may increase dramatically. This situation should be further investigated at subsequent design stages.

In a similar accident at sodium reactor, Na will burn at a rate of 1 to 3 t per hour. If 15 % of Na combustion products are carried over, release of ^{24}Na alone will exceed 10^5 Ci in four days (considering its decay) – and this is the minimum level, let alone ^{22}Na, ^{137}Cs and ^{131}I. Much sodium will burn out during this period. As a result, fuel will not be cooled any longer and will fail, which will cause a much greater release of radioactivity.

Given the worst weather conditions and disregarding precipitation from air, concentration of Na and its compounds in air at the distance of 10 km from the site may reach 10 mg/m^3, which is equivalent to 100÷1000 MPC in terms of Na chemical toxicity (for Na compounds MPC = 0.01 – 0.1 mg/m^3).

Thus, the analysis showed that release of radioactive and toxic materials in the accident involving failure of external barriers, will be three times greater in sodium reactors than in lead-cooled reactors because of sodium burning. This estimation does not take into account the consequences associated with loss of fuel cooling and its subsequent failure in the course of Na burning.

3. RESEARCH IN PHYSICS

Experimental and computational studies on the BREST-OD-300 reactor physics are largely aimed at:

— setting up a system of constants and software properly verified and certified by the Russian Regulatory Authority;
— demonstrating the degree of reliability and accuracy in calculating the main neutronic characteristics of the BREST-OD-300 core;
— proving the nuclear and radiation safety of the BREST-OD-300 facility and its on-site fuel cycle;
— adjusting and refining the design parameters of control systems, first and subsequent critical charges, refuelling programmes.

An international test model of the BREST-OD-300 reactor was developed for verification purposes. It appears sensible to have its characteristics analysed by as many Russian and foreign experts as possible with the use of various neutronic codes and nuclear data. The international scientific community is invited to take part in such testing. A test model for verification of transient and accident analysis codes is under development at present.

Validation of nuclear data and computer codes relies on the following integral experiments:

— measurement of removal cross-sections, determining the number of in-pile fast fissions, which allows judging the accuracy of description of the inelastic scattering process at 0.8–10.5 MeV;
— a series of critical experiments on spherical models with uranium and plutonium fuel and with a lead reflector of varied thickness;
— a series of experiments at the ROMB facility with “pancakes” of highly enriched uranium and lead,
— a series of experiments at BFS-1 and BFS-2 facilities in NRC “FEI”.

Of special importance among those is the work with the BFS facilities, which was carried out under a special extensive programme for experimental investigation of the BREST-OD-300 neutronics and for reactor modelling [5].

The assessed accuracy of calculating the main neutronic parameters of BREST-OD-300 is an integral numerical value which reflects the progress made in understanding the reactor physics. Such values are given in Table II with reference to certain key dates and in comparison with the tentative requirements. The year 1990, when macroscopic experiments were practically nonexistent, was taken as a starting point for the analysis. The recent situation (2002) is characterised by several good benchmarks and an experiment on a part-scope model of this reactor. As seen from the table, perceptible progress has been made in respect to many key parameters, with convergence of required and actual accuracies. According to current plans, at least the first phase of experiments on a full-scope BREST-OD-300 model at the BFS facility is to be completed in 2004. Assessments show that, given today's errors of experimental procedures, the required accuracy is likely to be attained for some parameters, such as criticality and power density distribution. Not so optimistic is the prospect for such parameters as the breeding ratio, some reactivity effects, and the worth of controls.

While developing the demonstration reactor BREST-OD-300, its designers realise that it is only pilot operation of the reactor that will allow reaching the required accuracy in calculating its physics, which is essential for description of reactivity variations in the course of establishing an equilibrium fuel composition and during subsequent operation in the equilibrium mode. Considering the similarity of fuel composition in the demonstration and commercial BREST reactors, the results of pilot operation may be assimilated in the commercial reactor design, using the perturbation theory methods.

Table II. The required and achieved accuracy in calculating the main neutronic characteristics of BREST-OD-300 reactor

Parameter	**Required**	**Achieved**		
		1990	**2002**	**2004**
K_{eff}	0.5%	2.5%	1.0%	+
Breeding ratio	0.02	0.06	0.04	?
CPS controls' worth	5%	30%	20%	?
Power density distribution	2%	5%	3%	+
Void effect of reactivity	0.2%	1.1%	0.4%	+
Doppler	10%	20%	15%	+

4. BREST-OD-300 FUEL TESTS IN BOR-60 REACTOR

In 1998-2001, a loop channel (or, in fact, a loop-in-channel construction) was designed and manufactured for testing BREST fuel rods in the BOR-60 reactor. This facility has a fuel assembly with BREST reactor fuel rods which differ from the real components only in length. The loop channel is fitted with all the components of the BREST reactor, such as fuel assembly, circulation circuit, oxygen monitoring and maintenance instruments, system for removal of impurities from the circuit, temperature monitoring and cladding failure detection devices. In the process of the loop channel manufacture, various technologies were tried out and perfected, including fabrication of U-Pu mononitride fuel [6], fabrication of the fuel cladding from EP-823 steel at the factory, lead injection, sealing, monitoring, the processes of fitting together the fuel assembly and the channel itself. Between January and May of 2002, the loop channel spent 2500 hours in the reactor and the fuel burnup reached 0.44%. Today, the channel and fuel rod materials are undergoing postoperational studies. Visual examination of the channel and the fuel rods after their cutting showed the steels to be free of corrosion. The design, manufacture and testing of the channel and fuel rods are described in detail in Ref. [7].

5. THERMAL-HYDRAULIC STUDY OF THE CORE

The work on the BREST-OD-300 design called for experimental studies on the thermal hydraulics of the core [8]. Considering the low coefficients of heat transfer in the lead coolant as compared to sodium (e.g. in BN reactors) and the lack of practical knowledge on the square fuel lattice employed in these reactors, investigations were carried out to determine the effect made on heat transfer coefficients by Peclet number (Pe), fuel rod pitch (s/d), spacer grids, radial and axial variations in power density, and by other factors typical of BREST reactors. Serious attention was paid in the studies to variations in the temperature of fuel rods in the regular lattice and of those located at the boundaries of the core regions with different fuel rod diameters and heat rates.

Experimental investigation of heat transfer coefficients and fuel rod temperature distributions was performed with the use of *thermal-hydraulic models* of the same design but differing in the mockup fuel rod pitch (s/d=1.46; 1.28 and 1.25), as well as in the absence or presence of spacer grids. The models were made as assemblies of 25 mockup fuel rods in a square lattice, placed in a rectangular shell. At the central mockup rod, of rotary design, surface temperature measurements were taken along its perimeter and length by means of micro-thermocouples caulked in the surface or moving along the rod. Coolant temperature was measured in all cells at the assembly outlet.

The *coolant was simulated* by a eutectic sodium-potassium alloy (22% Na + 78% K), with the Prandtl number numerically close to that of lead. Thereby close similarity of heat exchange processes at the “fuel rod – coolant” interface was provided, assuming certain “purity” of the coolants under consideration and absence of thermal-chemical phenomena on the heat exchange surface.

Thermal modelling of fuel rods (fissile material – uranium or plutonium mononitride, cladding – stainless steel, bond – lead) was fairly rigorous (with an accuracy of 5 %) using the fourth harmonic of temperature field expansion into Fourier series ($\kappa_0 = 4$), which is basic to the regular square fuel lattice. The experimental results are fully described in Ref. [8].

6. STEEL CORROSION TESTING

The problem of steel resistance to corrosion in lead is tackled with the use of a special lead coolant technology which allows forming protective oxide films on steel surfaces and keeping their thickness within optimal limits. This technology was first applied at facilities using Pb–Bi eutectic. The operating experience of these facilities was instrumental in choosing structural materials for the reactor under development, with this choice subsequently justified by long-term tests in lead under near-operational conditions in terms of lead temperature flow velocity. Steels of various classes were tested: e.g. austenitic, pearlitic and ferritic-martensitic steels with 9 to 12 % of Cr.

Apart from compatibility with liquid lead, the choice of materials was also guided by other criteria, such as high-temperature strength, manufacturability, resistance to radiation (for core materials) and to corrosion in high-parameter steam (for steam generator materials).

As a result, the structural materials recommended for the BREST-OD-300 reactor were assessed for corrosion resistance on the strength of 13500-hour tests [9].

With the optimal coolant oxygen regime identified, it is now one of the parameters involved in corrosion fatigue tests of candidate steels and their welds (tests have gone for 6,000 hours, of the 20,000 hours planned).

First studies have been carried out to test the steel for durability in liquid lead at 550°C. Liquid lead has not been found to have any significant effect on the long-term strength of the steels. Its reduction in lead is no greater than 10-13%.

The tested prototype pumps have proved their serviceability over a stretch of 3000 hours. Specifications were developed for the semifinished items required for the reactor, and their production batches have already been received. Welding methods have been developed for some reactor components.

Work on demonstrating the performance of the selected steels and their welds is being done today in the following main areas:

— study of steel corrodibility in liquid lead, lead vapour, at the liquid lead – inert gas interface, and in water vapour;
— investigation of mechanical properties of steels exposed to liquid lead;
— investigation of the irradiation effect on the structure, physical, mechanical and corrosion properties of steels in liquid lead;
— development of technologies for manufacture, bending and welding of semifinished products; quality assessment of the semifinished products and welds

7. COOLANT TECHNOLOGY JUSTIFICATION

The ever-increasing requirements for the safety and reliable performance of nuclear reactors spur up the search for new coolants which will have advantages over traditional fluids, such as water, sodium, etc.

One of such coolants is liquid lead. In its physics and chemistry, liquid lead is rather similar to the lead-bismuth eutectic. A great body of data has been amassed on the physical, chemical, thermal and other properties of this alloy. Procedures and experimental capabilities are available and are actually being used for justifying the choice of lead as a coolant for power reactors, of which BREST-OD-300 is the forerunner.

Lead dissolves many chemical elements and compounds, including some components of structural materials, with the ensuing risk of structural deterioration and loss of circuit integrity. Material dissolution (corrosion) may be effectively slowed down by protective iron and chromium oxide films on steel surfaces. Liquid lead reacts perceptibly with oxygen, giving rise to slag (containing oxides of the coolant itself, of structural steel components, etc.), which may deposit on the circuit surfaces, impairing its thermal-hydraulic characteristics.

Oxygen content in the coolant and in the circuit as a whole is a highly important factor in normal operation of the circuit. Oxygen excess leads to slagging, whereas its deficiency is likely to cause dissociation of protective oxide coats on structural materials and development of corrosion processes. Therefore successful operation of BREST-OD-300 will depend on properly controlled coolant quality, with the content of impurities kept at an optimal level (e.g. of oxygen, oxide compositions based on structural materials, etc.).

Provisions should be also made to prevent fouling of the gas plenum components and to clean the gas circulating in the reactor’s gas plenum from the products of lead evaporation, material corrosion, and other impurities.

All these problems are addressed in implementing the lead coolant technology. The term “coolant technology” implies a package of organisational and technical work, processes and their associated systems (facilities), all aimed at providing the required cleanness of the circuit and corrosion resistance of its structural components during construction, startup, maintenance and operation of an experimental rig or a full-fledged reactor facility.

Computations and experiments [10] resulted in the choice of the following lead coolant technologies accepted for further development:

- coolant and circuit treatment by hydrogen;
- coolant quality control by maintaining its oxygen content with the use of solid-phase oxidiser;
- removal of solid impurities from the coolant by means of filters;
- gas cleaning from suspended particles and aerosols by means of filters.

The information obtained to date from studies and experiments is sufficient for developing a draft procedure for lead coolant management.

8. FUEL CYCLE OF BREST-OD-300

The BREST-OD-300 project includes a nuclear power plant with a demonstration 300 MWe liquid metal reactor BREST, an on-site fuel cycle and a complex for treatment and storage of radioactive waste. The design studies undertaken to prove feasibility of BREST reactors of different power (600 and 1200 MWe) to be used in the prospective large-scale nuclear power, followed the same philosophy as that of the 300 MWe reactor.

The nuclear fuel cycle of BREST-OD-300 [11] affords practically unlimited availability of fuel resources for nuclear power, owing to recycling of U-Pu fuel with equilibrium composition (CBR~1), so that only small amounts of depleted or natural uranium need to be added to it. Moreover, such a cycle allows attaining radiation equivalence with allowance for migration. With radiation equivalence, waste sent to disposal, will have activity and nuclide composition such that the temperature and stability of material subject to disposal and the risk of nuclide migration, considering their biological hazard, are similar to those of natural uranium deposits.

The closed on-site fuel cycle of the BREST-OD-300 plant was designed to provide fabrication of 14 t of (U, Pu)N fuel per year, including ~3.5 t (U, Pu)N/year for BREST reactor and another 10.5 t (U, Pu)N/year for BN-800 under construction at the Beloyarsk site. The on-site facilities were also designed to provide fabrication of 49.1 t of (U, Pu)N fuel for the first cores of BREST and BN-800 reactors (17.6 t and 31.5 t, respectively). The on-site fuel cycle consists of the following fabrication-related facilities:

— production of Pu mononitride;
— FA cutting and stripping of fuel rods;
— reprocessing;
— preparation of press powder and fuel fabrication;
— preparation of claddings and fuel rod components;
— fabrication of fuel rods;
— fabrication of fuel assemblies.

In the BREST-OD-300 project, the emphasis is on engineering rather than administrative nonproliferation provisions. This is proved by the following design features:

— there is no blanket in BREST-OD-300. Any specially fabricated Pu-breeding fuel assembly placed in the core under equilibrium operation, will cause introduction of considerable negative reactivity, which will be inevitably detected during reactor startup after refuelling;
— the BREST fuel cycle does not include shipment of spent fuel assemblies to a reprocessing plant. Spent fuel assemblies are to be sent after a year of cooling to on-site cycle facilities via a transport corridor connecting the latter to the reactor section. This helps avoid all risks and costs associated with fuel shipment for reprocessing and obviates the need for the associated transportation and handling gear;
— BREST fuel - before or after reprocessing - cannot be used for production of nuclear charges without a proper separation facility. A crucial requirement for reprocessing technique is to keep uranium and plutonium inseparable and their proportion in fuel constant at all reprocessing stages;
— fuel reprocessing and fabrication takes place in unattended heavily shielded cells;
— reprocessed fuel contains up to 1% of fission products, which simplifies safeguarding of nuclear material. (This feature of BREST fuel is sometimes referred to as “inherent safeguards”).

9. EXPERIMENTAL CAPABILITIES FOR R&D IN SUPPORT OF THE BREST TECHNOLOGY

The work on design of the demonstration plant with a 300 MW lead-cooled fast reactor (BREST-OD-300) and on-site fuel cycle at Beloyarsk has progressed to a level where the experimental studies for design validation need to be expanded.

In 1999-2002, the participating organisations (NIKIET, FEI, VNIINM, NIIAR, VTI, CNIITMASH, CNII KM "Prometei", OKB "Gidropress", "Gidromash, and others) launched project justification studies using the available experimental facilities and focusing on the flowing areas [12]:

— corrosion studies and qualification of structural materials for the core and reactor internals (FEI, CNII KM "Prometei", NIKIET, CNIITMASH);
— neutronic studies (FEI);
— thermal-hydraulic studies (FEI);
— research in coolant technology (FEI);
— in-pile tests (NIIAR, Sverdlovsk Branch of NIKIET)

10. CONCLUSION

A nuclear generating complex has been designed, including a demonstration plant with BREST-OD-300 reactor, on-site nuclear fuel cycle and radwaste treatment facilities, to be built on the Beloyarsk NPP site. The experiments planned for the complex are expected to confirm the feasibility of building a large-scale nuclear power industry based on fast reactors. Full-scope R&D should be continued to have the design comprehensively validated.

Ample design documentation has been produced to obtain a license for the site, to start review procedures (governmental, environmental, regulatory, etc.) and design refinements so that a construction license may be obtained in a 3 or 4 years' time.

The cost of the BREST Project, comprising the plant proper, the fuel cycle and the radwaste treatment facility to cater for the BREST-OD-300 and the BN-800 under construction on the same Beloyarsk site, is estimated at US$ 825 million, including:

Capital cost of the NPP	285 M$
Capital cost of the NFC and RW complex	355 M$
Project R&D	185 M$

The nuclear generating complex built to this design will have nuclear fuel delivered to it only once, to make the first core, while its radioactive waste, treated and cooled for ~ 150 years, will have a radiotoxicity equivalent to that of mined uranium and therefore may be buried without upsetting the natural radiation balance.

11. REFERENCES

[1] Ministry of the Russian Federation for Atomic Energy, Strategy of Nuclear Power Development in Russia in the First Half of the 21st Century, Summary, Minatom of Russia, Moscow (2000).

[2] ADAMOV, E.O., et al., “Nuclear power development based on new concepts of nuclear reactors and associated fuel cycle. President Putin’s Initiative” (Proc. 11th Int. Conf. on Nucl. Engng, ICONE-11, 2003), JSME/ASME, Shinjuku, Tokyo, Japan (2003) (CD-ROM file ICONE11-36412).
[3] GABARAEV, B.A., “Development of a BREST-OD-300 NPP with an on-site fuel cycle for the Beloyarsk NPP. Implementation of the Initiative by Russian Federation President V.V. Putin”, ibid., (CD-ROM file ICONE11-36410).
[4] ORLOV, V.V., et al., “Deterministic safety of BREST reactors”, ibid., (CD-ROM file ICONE11-36415).
[5] ORLOV, V.V., et al., “Experimental and Calculation Investigations of Neutron-Physical Characteristics of BREST-OD-300 Reactor”, ibid., (CD-ROM file ICONE11-36406).
[6] VATULIN, A.V., et al., “Mononitride uranium-plutonium fuel of fast lead-cooled reactors”, ibid., (CD-ROM file ICONE11-36414).
[7] LEONOV, V.N., et al., “Pre- and in-pile tests of fuel element mock-ups for the BREST-OD-300 in the independent lead-cooled channel of the BOR-60 reactor”, ibid., (CD-ROM file ICONE11-36409).
[8] SMIRNOV, V.P., et al., “Thermohydraulic research for the core of the BREST-OD-300 reactor”, ibid., (CD-ROM file ICONE11-36407).
[9] ABRAMOV, V.Y., et al., “Corrosion and mechanical properties of BREST-OD-300 reactor structural materials”, ibid., (CD-ROM file ICONE11-36413).
[10] ORLOV, Y.I., et al., “Validation of the lead coolant technology for BREST reactors”, ibid., (CD-ROM file ICONE11-36408).
[11] ORLOV, V.V., et al., “Fuel cycle of BREST reactors. Solution of the radwaste and nonproliferation problems”, ibid., (CD-ROM file ICONE11-36405).
[12] BEZZUBTSEV, V.S., et al., “Experimental base, available and under construction, for R&D aimed at the BREST reactor design”, ibid., (CD-ROM file ICONE11-36411).

CAREM CONCEPT: A COST EFFECTIVE INNOVATIVE LWR FOR SMALL AND MEDIUM UTILITIES

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Abstract. CAREM concept is based in a cost effective primary system configuration. A proper safety balance by design is assured to avoid jeopardizing reactor economic competitiveness. An innovative methodology to perform or assist reactor design, balancing safety and economics at the conceptual stage is used in CAREM project. The key to this integral methodology is to take into account safety aspects in an optimization design process where the design variables are balanced in order to obtain a better figure of merit related with reactor economic performance. Proliferation resistance is another important aspect taken into account in CAREM concept. CAREM design includes isolated Material Balance Areas that contain all the fissile material and are remotely checked. But proliferation resistance is useful when all the fuel cycle is included, so initial enrichment and fuel cycle length is optimized considering the different safeguard costs. In this paper CAREM concept approach to obtain an innovative small nuclear power plant with a cost effective economic, safety and proliferation resistance balance is presented.

1. INTRODUCTION

There is an important need of small nuclear power plants suitable for developing countries and small or medium developed countries utilities. CAREM concept is conceived to offer nuclear options to this market, with 1000 U$S/KWh overnight cost for a 300 MWe nuclear power plant.

In order to be part of the new power supply demand the design of future nuclear power plants must guarantee their good performance considering such issues as economy, safety, and simplicity during construction, operation and maintenance. They should be competitive when comparing these aspects with those of the present nuclear and non-nuclear power plants. It is becoming evident that classical methodologies to perform nuclear reactor design must be reviewed and new ones developed aiming at achieving this competitiveness. It is important to carry out this process with a global approach, contemplating design feedback effects between all the systems and involved areas.

Reactor design is an intrinsically complex task, due to the quantity of parameters whose dimensions have to be determined and the existing relations between them. Moreover, there are often lots of alternative ways to accomplish the same results. Compromise issues are inherent to the problem when considering safety and economics together; this makes the designers' task to be quite interesting.

In the last few years, in some advanced reactor designs, some design parameters have been optimized to minimize the plant costs when it works in its nominal point, though without taking into account its behavior when accidents occur. Moreover, in many cases safety aspects were evaluated once all the main systems and engineered safety feature designs were defined, being then added as a "patch"; and higher safety levels usually implied a significant increase in plant costs. Problems have also appeared concerning conflicts between new requirements, classical design concepts and operational demands –in view of the economic reality–, which were previously often ignored. These new design requirements for advanced reactors, when not properly handled, cause a loss of competitiveness. It turns out that the problem at hand is to accomplish cost-efficiently the safety internalization process. Therefore, a new integral design approach must be developed to fulfil the market conditions that require an economical and safe electric power production.

A proper safety balance by design should be assured to avoid jeopardizing reactor economic competitiveness. An innovative methodology to perform or assist reactor design, balancing safety and economics at the conceptual stage is used in CAREM project. The key to this integral methodology is to take into account safety aspects in an optimisation design process where the design variables are balanced in order to obtain a better figure of merit related with reactor economic performance [1,2,3,4,5]. The design parameter effect on characteristic or critical safety variables, chosen from the reactor behaviour during accidents (safety performance indicators), is synthesised in Design Maps.

This methodology allows to balance and optimise reactor itself and safety system in an early engineering stage in order to internalise cost-efficiently safety issues, based on the defence in depth approach, considering appropriate conservative assumptions and safety margins. Therefore, a balance between reactor inherent capability and safety systems to cope with the postulated initiating events can be achieved. Finally this balanced design prevents that the search for economic performance should cause less safe reactors and, likewise, guarantees the design competitiveness in spite of unavoidable safety costs.

Proliferation resistance is another important aspect taken into account in CAREM concept. CAREM design includes isolated Material Balance Areas that contain all the fissile material and are remotely checked. But proliferation resistance is useful when all the fuel cycle is included, so initial enrichment and fuel cycle length is optimized considering the different safeguard costs.

In this paper CAREM concept approach to obtain an innovative small nuclear power plant with a cost effective economic, safety and proliferation resistance balance is presented.

2. CAREM NUCLEAR POWER PLANT

The main characteristics of CAREM nuclear power plant [6], are described in this section.

The CAREM design is based on an integrated light water reactor with slightly enriched uranium fuel. It is an indirect cycle reactor with some distinctive features that greatly simplify the design and also contribute to a high level of safety. The main design characteristics are:

— Integrated primary cooling system
— Primary cooling by natural or assisted circulation depending on the module power
— Self-pressurized
— Passive safety systems

The primary cooling system is integrated. The reactor pressure vessel (RPV) includes the core, the steam generators, the whole primary coolant and the absorber rod drive mechanisms.

For low power modules (below 150 MWe), the flow rate in the reactor primary systems is achieved by natural circulation. Figure 1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After been heated the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing the adequate flow rate in the core in order to have the sufficient thermal margin to critical phenomena. Reactor coolant natural circulation is produced by the location of the steam generators above the core. Coolant acts also as neutron moderator.

For high power modules (over 150Mwe) pumps are used to achieve the flow rate needed to operate at full power.

The steam generators are the "once-through" helical tubes type. In these generators the flows of the primary and secondary systems circulate in countercurrent flow. The secondary fluid circulates upwards within the tubes. It flows into the tubes as liquid-water and it reaches the exit as overheated vapor.

Self-pressurization of the primary system in the steam dome is the result of the liquid-vapor equilibrium. The large volume of the integral pressuriser also contributes to the damping of eventual pressure perturbations. Due to self-pressurisation, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. Heaters and sprinkles typical of conventional PWR's are thus eliminated.

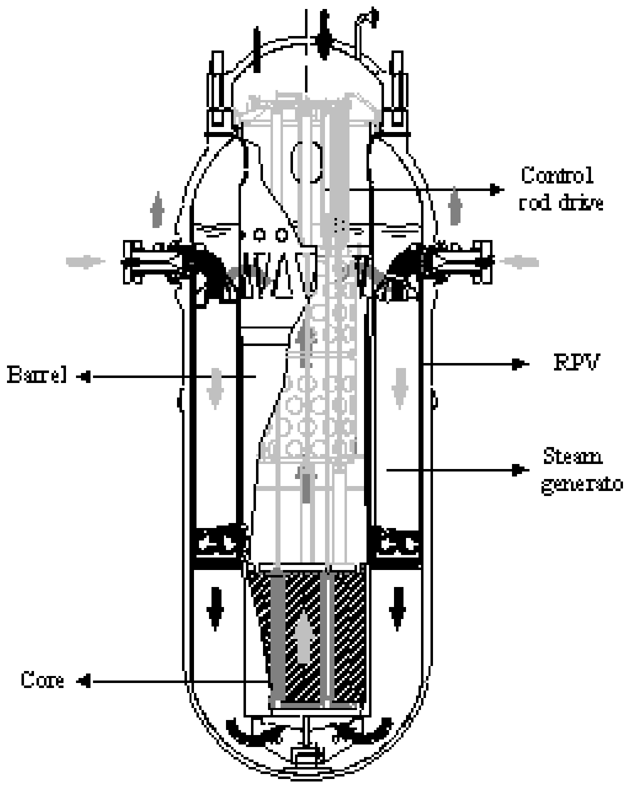

FIG. 1. Reactor pressure vessel – Primary system.

The fuel assembly components are similar to those of a conventional PWR design.

Core reactivity is controlled by Gd_2O_3 as burnable poison in specific fuel rods and movable silver–indium-cadmium absorber rods. The control rods drives are hydraulic and are placed inside the RPV. Chemical compounds are not used for reactivity control during normal operation.

The design of the security systems fulfils the requirements of the regulations of the nuclear industry as for redundancy, independence, physical separation, diversification and failure into a safe state.

CAREM safety systems must guarantee no need of active actions to mitigate accidents for a long period.

CAREM has two different and independent shutdown systems. These systems are designed to shutdown and to maintain the reactor core sub-critical. They are activated by the protection system reactor. The first system is designed to shutdown the core reactor by dropping neutron-absorbing elements into the core by the action of gravity. The second shutdown system is based on the injection of borated water to the core, also by the action of gravity.

The residual heat of the core, in station blackout, is removed by passive principles (natural convection) through the residual heat removal system. This system transfers this energy to the pressure suppression pool.

CAREM has an emergency injection system to prevent core exposure in case of loss of coolant accident (LOCA). This system assures the correct refrigeration of core reactor without electric power supply.

The RPV integrity is additionally ensured by three safety relief valves. They protect the RPV against overpressures and each valve has 100% of the necessary relief capacity.

CAREM has a containment isolation -pressure-suppression type- to retain the eventual liberation of radio-active materials. Its design is such that after having begun any unlike accident with loss of coolant, and without any external action, the pressure inside stays below the design pressure.

In the Figure 2 an outline of the containment and safety systems are shown.

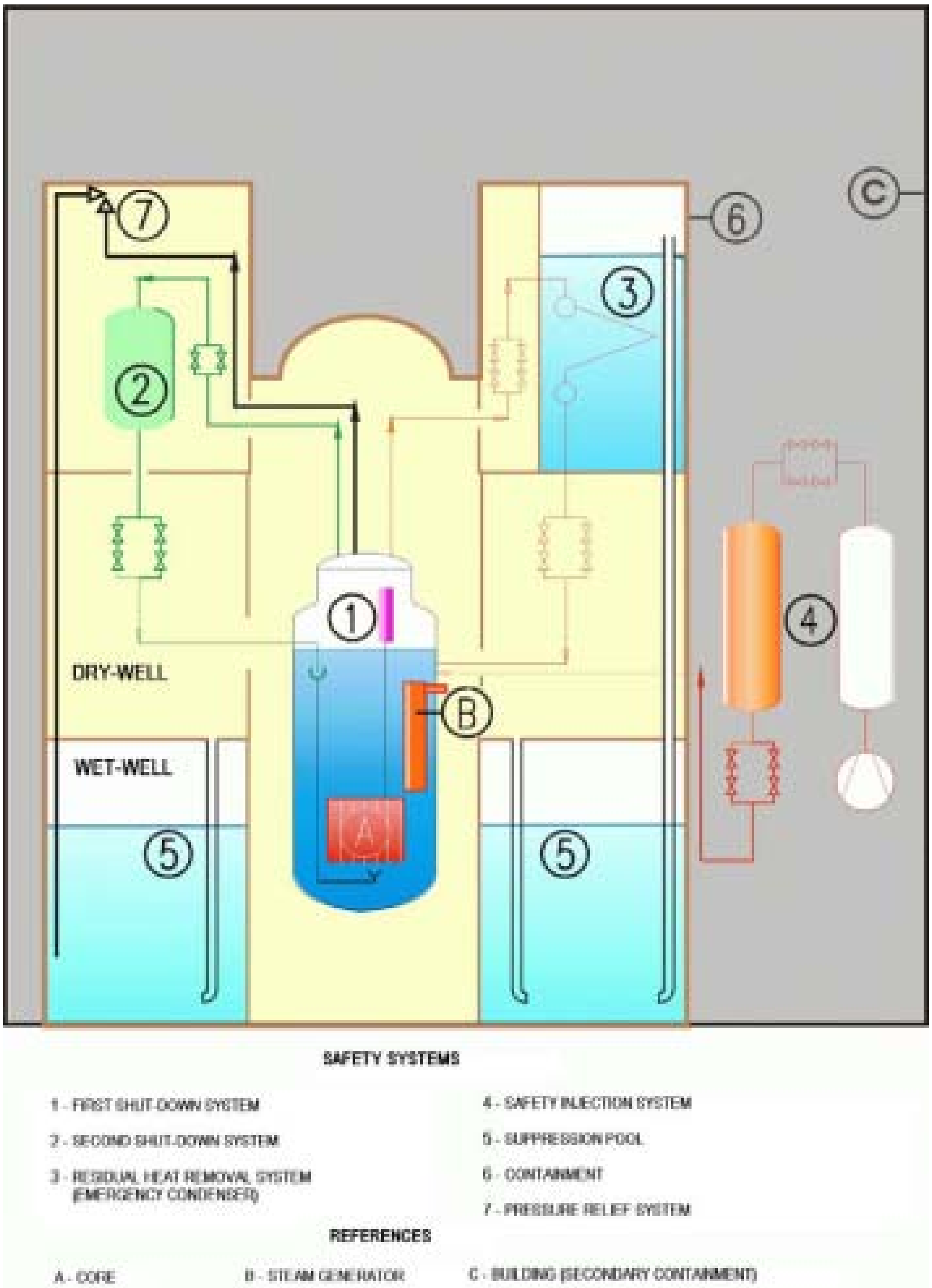

FIG. 2. Containment and safety systems.

3. CAREM AND THE COST EFFECTIVE INTERNALIZATION OF NUCLEAR SAFETY

Reactor design is an intrinsically complex task, due to the quantity of parameters whose dimensions have to be determined and the existing relations between them. At the engineering conceptual stage, quantifying the influence of mechanical, thermal-hydraulic and neutronic parameters on reactor costs is of interest. A breakdown of the main items that affect costs must be performed with the purpose of finding a unit cost for the generated energy, a figure of merit of the alternative designs.

Under the program of CAREM integral type reactor, a computational tool is been developed to perform the above mentioned tasks as support to the design team during the conceptual stage. This code, called IREP –Integrated Reactor Evaluation Program–, makes the necessary internal iterations to obtain a coherent set of design and operational parameters that define a reactor, considering the main feedback existing between these parameters. This code also allows the designers to optimize economically the most important parameters of the core, primary, safety systems and secondary systems, in order to reduce the cost of electricity generation.

3.1. IREP code

IREP performs the neutronic, thermal hydraulic, mechanical, economical and nuclear safety evaluation of the reactor, and gives the leveled electricity generation costs as a main result, providing all the technical outcomes of the different areas as well. This code allows carrying out an optimization of the most influential parameters in the generation cost of the electricity generated by the designed reactor. Before performing the optimization, a series of main engineering decisions such as the reactor power are given to the steady-state calculation routines. At the beginning, a set of the main design parameters that correspond to an initial design are introduced to these routines, which have the mechanical, thermal-hydraulic, neutronic and economic models to calculate the plant. The results, which include the figure of merit, namely the generation cost, enter the optimizer routine. Design restrictions, are verified while this routine looks for a more economical design. The new set of design parameters replaces the previous one and the process continues iteratively.

3.2. IREP safety aspect

Although the current methodologies, classical or more advanced ones, like a steady state optimization, fulfil the requirements of design relative to safety, the lack of balance between economy and safety is evident. It is necessary that economy and safety should be evaluated together in the conceptual design stage, to balance properly these two fundamental aspects of design. It is important to perform this process with a global approach, contemplating the design feedback between all the systems and involved areas. Safety aspects are part of the most important contributors to costs, hence they must be considered in an efficient way. As other authors have already noticed, the new approach must consider new methods for cost-benefit and ALARA analyses, employing modern PSA techniques and fulfilling basic safety requirements instead of overly detailed prescriptions, with realistic models and assumptions.

The conceptual global design process in order to design the reactor to be safe and competitive, performing an integral optimization of the design parameters, can be resumed in the following stages considering the above analysis:

(a) *Preliminary conceptual design and qualitative optimization based on designers' judgment.* Stage based on designers' expertise and research results, recognizing alternatives that aim to simplify the design and to reduce initiating events and diminish their incidence, among other design goals. Different alternatives for safety and process systems are proposed at this stage, for being evaluated in the next one. Thus, the design basis is now obtained.

(b) *Integrated conceptual design and quantitative optimization.* This second stage consists in an integral design optimization process in order to improve a figure of merit. To perform this, neutronic, thermal-hydraulic, mechanical, safety and economical dimensioning modules are required. Safety ones are employed to simulate the plant performance in steady state and in transients or accidents and to characterize it by means of safety performance indicators. This evaluation is performed for each set of parameters that defines a possible reactor design that may be found during the optimization process. Safety goals determined by regulators and designers are embodied in practical quantitative safety targets. They are applied as limits to the selected safety performance indicators and therefore considered as restrictions on the design parameters. Then, the economic figure of merit is calculated given the main design parameter values. Finally, the optimization gives a new set of parameters improving the value of the figure of merit. This stage is repeated until the design converges.

(c) *Final conceptual design stage based on experts' judgement.* Evaluating the alternatives results, the best design options are chosen. Eventually, feedback to previous steps will be necessary.

3.3. Optimization methodology: general concepts

In order to face the posed design optimization problem, an objective is selected, a feature that is being analyzed and should be reached with the design. It is a result of the design parameters, which witnesses how good or bad a design is, in relation to the proposed goal. It is called figure of merit. Aiming at designing competitive nuclear power plants, adopted strategies may include the reduction of capital costs or other economic figures of merit. Several results of the design process can be selected as figure of merit for economical optimization. They are typically electricity generation cost, cost of investment by power unit ($/kw), total investment cost (releasing power as a parameter to optimize) and net present value of the project (assuming a known price of sale of the energy unit).

To verify reactor safety criteria fulfilment, the concept of safety performance indicators is introduced, also known as response functionals or observable variables. Each one of these variables is chosen in order to characterize and represent reactor safety levels or reactor degree of exigency during an accidental sequence. The idea is that for each accidental sequence, one or more indicators or observable variables can be defined. It is important to identify all the observable variables, which can be critical for assuring the reactor safety in every transient and postulated initiating events, because the success of the design will depend on the restrictions applied to them. Probabilistic limits are also supported by the methodology, included as further safety indicators. An example of a safety indicator is the time that the water level inside the RPV in an integral-type PWR takes to reach the core top during a loss of coolant accident (LOCA). The Minimal Critical Power Ratio reached during a reactivity insertion accident is another one. Probabilistic safety indicators, such as the core damage probability, can also be considered. Operational performance indicators are studied in reference.

There are also restrictions, which are limits that a particular design must fulfil and are applied to the design parameters as well as to the safety indicators. It is evident that the value of each safety indicator will be function of the design parameters. During the optimization process developed, while looking for an appropriate set of design parameters that optimizes a given figure of merit related with cost, safety indicators are compared with imposed limits. Should any of these limits be violated, the direction of the design parameters movement is changed in order to keep the reactor safe enough. Therefore, the safety indicators will be used to evaluate the safety degree and to determine the direction the design parameters must move towards, within the general scheme of optimization, as explained below.

Besides verifying safety criteria, the safety indicators can be also considered as a figure of merit to be improved instead of a cost related one. Cost-related or other design restrictions can either be considered or not, depending on the designers' choice. For instance, this could be used to find a feasible design (one that does not violate any restriction) when some safety restrictions are being violated, for a posterior economic optimization inside the feasible design region. Other uses would be to search the safest design alternative for a given generation cost or the "safest limited-budget design". The safety criteria fulfilment could be verified after these ALARA-like optimizations take place.

Considering then that the parameters dimensioning influences both the figure of merit and the safety indicators limited by restrictions, the concept of Design Map is reached. A Design Map is a representation of the safety indicator dependence in order to translate to the design parameters the restriction applied to the safety indicator.

The optimisation method works properly for the optimisation process performed either by calculating the safety indicators with the *online* approach (calling the calculation models for specific points as the optimisation takes place) or for those cases in which they are obtained reading their values from a design map obtained *a priori*. These safety indicators must be evaluated for the numerical calculation of their gradients and when a new restriction is violated, in order to reduce the parameter jump vector. It can be seen that it is faster to perform an optimisation by means of design maps obtained a priori in those cases where the calculation model is too slow to be called so many times in each optimisation step. For the cases in which the speed of execution of the calculation model is not too significant, there are no disadvantages of doing it online. A diagram of the whole calculation paths in the global design process is shown in Figure 3. In the bottom loop, an unrestricted optimisation path can be observed. In the upper part, the verification path for restriction fulfilment by means of the design maps is shown.

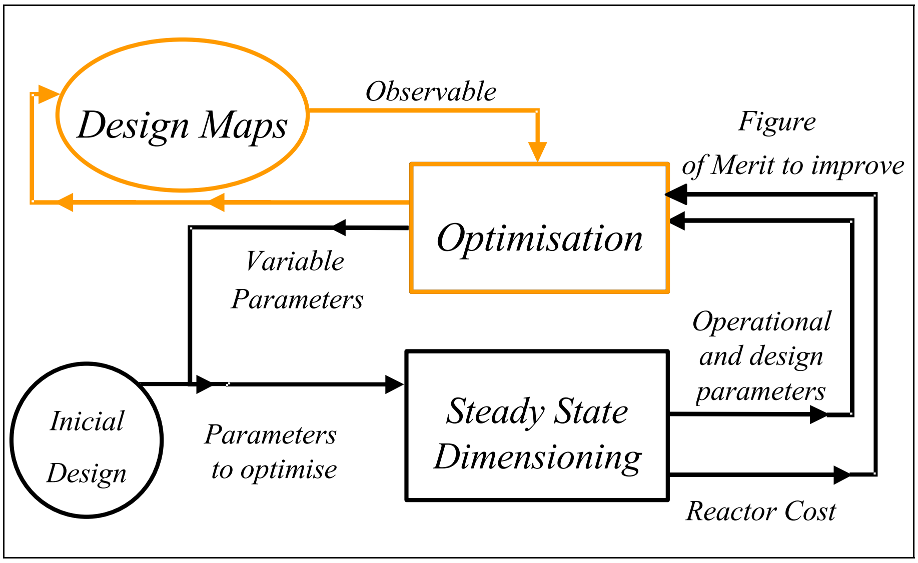

FIG. 3. Calculation diagram. Figure of merit: reactor cost.

4. CAREM AND A COST EFFECTIVE PROLIFERATION RESISTANCE

4.1. Safeguard implementation

Two isolated Material Balance Areas (MBA) for irradiated fuel are included in CAREM concept approach in order to facilitate safeguard implementation and reduce safeguard costs. One is the pressure vessel and the other is the spent fuel pool. This two MBAs have all the irradiated fuel and allow integrity check by remote systems. During reactor operation there is no physical way to get access to these fissile material.

To increase proliferation resistance all the refueling tasks will be developed in the reactor hall, which is designed to allow remote monitoring of all nuclear material handling. The entrance-exit and the interfaces have been designed to allow the counting of the items during they movement.

This approach allows a reduction of safeguard costs of about 10 to 20%.

4.2. Long life core

Many new designs propose the use of long life cores. This usually means the use of one zone core refuelling, higher enrichment and lower power density.

In order to obtain the safeguard effort associated all the fuel cycle needs to be considered. The result depends on the time schedule and the significant quantities.

The use of higher enrichment increase proliferation resistance by increasing burn-up and refuelling time, but reduce proliferation resistance due to the increase in ^{235}U in the first core and each refuelling.

The core life length could be optimized selecting for instance a common variable that should be applied to all the steps of the fuel cycle. The use of a discount rate is necessary as different steps occur at different time.

For a first analysis let us consider one main assumption: include all the steps and costs related to fuel for a one-trough fuel cycle until the fuel element is taken away from the reactor. Let us also consider one main constrain: the higher enrichment allowed is 20%.

The proliferation risk sources associated with the Fuel Cycle Front End are the conversion plants, enrichment plants and fuel manufacture plants. In these cases the safeguard cost depends on time and the quantities involved.

The proliferation risk sources associated with the reactor operation are the reactor fuel and the spent fuel. In these cases the safeguard cost depends also on burn-up and refuelling time.

Then the proliferation risk calculation should include the amortisation of the first core safeguard cost, the refuelling fuel safeguard cost, the reactor surveillance system safeguard cost and the irradiated spent fuel safeguard cost. This result could be expressed in levelled mills/KWh.

The influence of the use of different enrichment on CAREM prototype, CAREM-25, was analysed. Figure 4 shows the different safeguard cost for two refuelling zones, a given power density and different core enrichment. Figure 5 shows the fuel and safeguard cost of CAREM-25 for one and two refuelling zones, different power densities and different times between refuelling (i.e. different enrichment). An important result is that safeguard costs are very small compared against fuel costs and they do not significantly change the optimum enrichment. Long life cores were considered but the economical evaluation shown that for CAREM-25 design the optimum enrichment is below 5%.

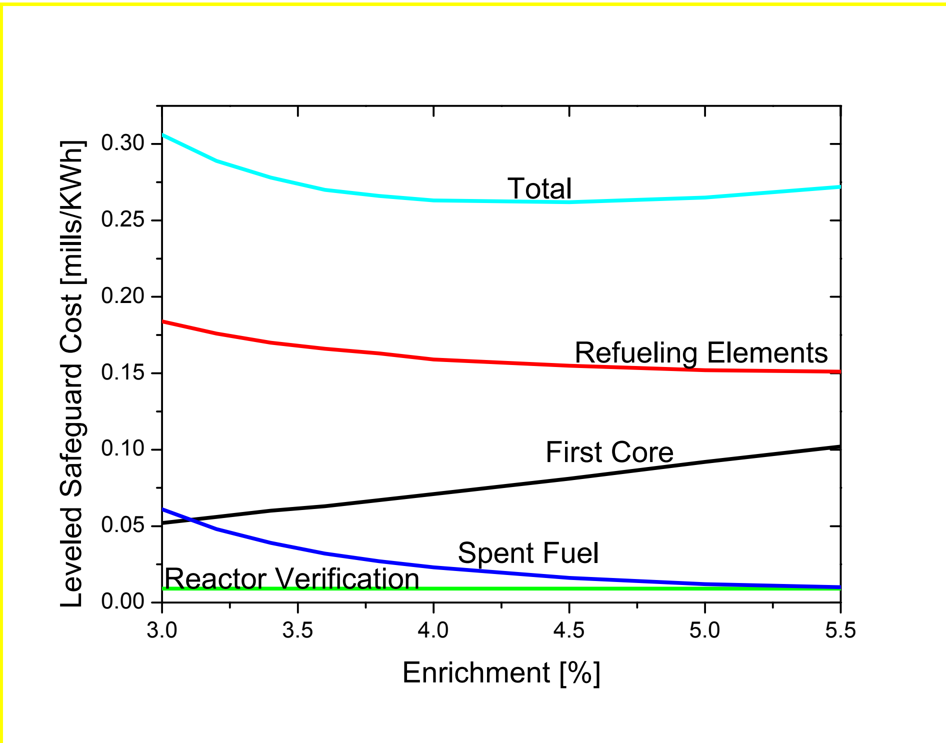

FIG. 4. CAREM-25 safeguard cost.

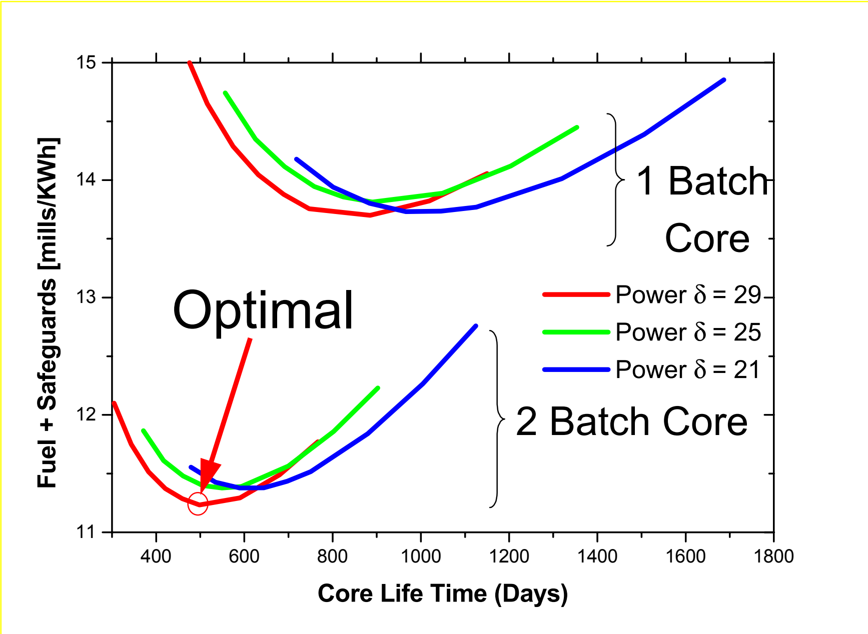

FIG. 5. CAREM-25 fuel and safeguard cost.

5. CONCLUSIONS

In this paper CAREM concept approach to obtain an innovative small nuclear power plant with a cost effective economic, safety and proliferation resistance balance is presented.

The present work presents a methodology to balance safety and economy of a nuclear power plant, aiming at achieving an efficient internalization of the external costs. One of the main outcomes is that it is possible to optimize a reactor design internalizing its safety costs efficiently. This process tends to cost reduction, a greater simplicity and a better strategy for prevention and mitigation. All of this is performed by integrating safety evaluation with neutronic, thermal-hydraulic and mechanical calculations in the design optimization. This methodology provides the instruments necessary to be able to guarantee that the adopted criteria for reactor safety (restrictions applied to safety-related performance indicators) are verified in each one of the optimization steps toward optimal cost.

Moreover, a relevant issue is that the present methodology allows one to incorporate reactor dynamic response during transients or accidents in an early engineering stage for design parameter integral optimization, by using safety design maps. This is done through new rules for neutronic, thermal-hydraulic and mechanical calculations –additional to those necessary for steady state dimensioning.

This is a promising methodology for equalizing and optimizing reactor and safety systems design in an early engineering stage. Therefore, a balance between reactor inherent capability and safety systems to cope with the postulated initiating events can be achieved. This equilibrium prevents the search for economic performance from causing less safe reactors and, likewise, guarantees the design competitiveness in spite of the unavoidable safety costs. Furthermore, by means of this methodology a simplified design can be obtained, compared to the resultant complexity when these concepts are introduced in a later engineering stage.

The present methodology has been implemented in a computational tool called IREP 3 and is being tested to balance the inherent safety in integral pressurized water reactors versus safety system capability to cope with LOCA and/or LOHS sequences. Likewise, this methodology may also be used to balance different but complementary safety systems to withstand a given initiating event. Moreover, it can be used to decide the best way to perform a safety function such as depressurizing the reactor during a small LOCA by means of an Automatic Depressurization System or by a Residual Heat Removal System.

A further application is the evaluation of the additional costs of higher safety levels, or those due to the uncertainties in the limits applied to safety performance indicators.

Another advantage is that in case of need, one of the safety performance indicators can be used as figure of merit to be improved. This situation could occur if a safety related variable violates a restriction in a given reactor design. Cost could, or could not, be considered as a new observable subject to restrictions in the same way as the rest of the safety indicators are.

In addition, the developed methodology offers the possibility of handling probabilistic limits to avoid the occurrence of non-wished events, such as core melt probability. Moreover, the uncertainty treatment can also be handled, considering both the uncertainties in the design parameters (and their effects on the costs and on the safety performance indicators) and those due to the models used by the code. These objectives will constitute the next development steps.

Another derived issue of this methodology is that it may have two main effects: faster licensing processes and flexibility for enhanced designs. Both of them will assist the nuclear industry to satisfy society's main demands: safety and cost-effectiveness.

It is important to mention that this methodology does not replace the judgement of experts and detailed accident simulations must still be done in order to verify reactor safety. Finally, a great deal of work remains to be done in order to explore and to make concrete the potential benefits of the methodology.

Proliferation resistance is another important aspect taken into account in CAREM concept. CAREM design includes isolated Material Balance Areas that contain all the fissile material and are remotely checked. But proliferation resistance is useful when all the fuel cycle is included, so initial enrichment and fuel cycle length is optimized considering the different safeguard costs. Long life cores were considered but the economical evaluation shown that for CAREM-25 design the optimum enrichment is around 4%.

6. REFERENCES

[1] CAREM: Nuclear Safety Internalized Cost-Effectively from the Concept Genesis. P. Grinblat, M. Giménez and M. Schlamp, International Congress on Advanced Nuclear Power Plants, ICAPP03, Córdoba, Spain, May 2003 (in Proceedings).

[2] Safety aspects in reactor conceptual design: a cost-effective internalization, M. Giménez, P. Grinblat and M. Schlamp. Twenty-Second Nuclear Simulation Symposium, Nuclear Power Reactors: Step Into the Future, Canadian Nuclear Society, Plenary Session, Ottawa, November 2002 (in Proccedings).

[3] Internalization of nuclear safety in optimized conceptual design of reactors, M. Giménez, P. Grinblat, M. Schlamp, IAEA, Technical Meeting on The Efficient Internalization of the External Costs of Nuclear Power, Vienna, November de 2001.

[4] A new cost-effective methodology for nuclear reactor conceptual design, M. Giménez, P. Grinblat and M. Schlamp, report IAEA under codification, April 2002. IAEA contractual service agreement No 097217, 2002.
[5] A new cost-effective methodology for nuclear reactor conceptual design: case study for loss of coolant accidents, M. Giménez, P. Grinblat and M. Schlamp, report IAEA under codification, April 2002. IAEA contractual service agreement No 097217, 2002.
[6] CAREM: un Reactor Avanzado Argentino. Delmastro,D.; Gómez de Soler, S.; Ramilo, L.; Mazzi, R. Symposium on Nuclear Energy: Challenge and Realities in Latin America. Buenos Aires. 2001.

THE IRIS REACTOR DESIGN
AN INTERNATIONAL COOPERATIVE PROJECT AND THE BRAZILIAN PARTICIPATION

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Abstract. IRIS, the International Reactor Innovative and Secure, is an Integrated Primary System Reactor (IPSR) with innovative features that can meet most of the requirements for the next generation of nuclear energy systems. The IRIS project is being conducted as an international development program in a collaborative approach and management openness. This initiative has found a very positive response around the world and the IRIS team grew from the initial four members from two countries to the present number of 22 organizations from nine countries. By the end of 2001, the Brazilian Nuclear Commission, CNEN, signed a collective agreement with Westinghouse Electric Company to officially participate in the development of IRIS. The IPSR concept of IRIS is characterized by the inclusion of the entire primary system within a single pressure vessel, including the steam generators and pressurizer. The CNEN team is responsible for the internal pressurizer design and for the review of RELAP5 input file. The design tasks for the pressurizer have included steam-water volume sizing, pressurizer to vessel physical separation by an internal thermal insulation, surge connections dimensioning and transient analyses. This paper presents a very brief description of IRIS, and a summary of CNEN activities in this project.

1. INTRODUCTION

An international collaborative initiative to develop an Integrated Primary System Reactor (IPSR) has created a consortium of industry, laboratory, university and utility establishments, led by Westinghouse to design the so called IRIS - International Reactor Innovative and Secure. Its concept is solidly based on proven LWR technology, but creatively put together with innovative features that can meet most of the requirements considered in the Generation IV Roadmap Study. The IRIS reactor has very good characteristics concerning economics, proliferation resistance, enhanced safety, modular deployment, innovative waste management and fuel utilization. IRIS has been selected as one of the International Near Term Deployable (INTD) reactors, within the Generation IV International Forum (GIF) [1].

The collaborative approach and management openness of this initiative has found a very positive response around the world and the IRIS team grew from the initial four members from two countries to the present number of 22 organizations from nine countries. By the end of 2001, the Brazilian Nuclear Commission, CNEN, signed a collective agreement with Westinghouse Electric Company to officially participate in the development of IRIS.

One of the challenges was the development of the IRIS pressurizer, which is located within the reactor vessel head and has no spray system, thus raising some interesting technical issues. The CNEN team has been assigned the responsibility for the internal pressurizer design and for the independent review of RELAP 5 input file.

The design tasks for this mission include steam-water volume sizing, design of the physical separation between pressurizer and reactor, surge connections dimensioning, honeycomb thermal insulation development, and assessment of pressurizer dynamics fitness to deal with operational transients and abnormal conditions as well. Related control issues like "Power – Temperature Program", and "Pressurizer Level Program" are also addressed by the CNEN team.

This paper, besides a very brief description of IRIS, will present a summary of CNEN activities in this project, concentrating on some of the more recent work dealing with operational transient analysis of the pressurizer behaviour. Finally, some of the management approach to such a large, world-wide design work force will also be addressed.

2. IRIS CONCEPTUAL DESCRIPTION

IRIS is a modular, integral, light water cooled, medium power [335 MW(e)/module] reactor which addresses the requirements of proliferation resistance, enhanced safety, improved economics and waste reduction. The 6.78m outside diameter by 21.4m in height IRIS integral vessel houses the reactor core, its support structures, upper internals, eight steam generators, internal shields, pressurizer and heaters, and eight reactor coolant pumps (Fig.1). Hot coolant rising from the reactor core to the top of the vessel is pumped into the steam generators annulus. The technical characteristics of IRIS are discussed in detail in references [2-7]. Its "safety by design" approach, where accidents are "designed out" to the maximum extent possible, instead of engineering to cope with their consequences is presented in [8].

Three most innovative features, which characterize IRIS project, are:

- Safety by design;
- Optimized maintenance; and,
- Long core life.

2.1. Safety by design

IRIS design takes advantage of the integral configuration to improve safety. Its configuration physically eliminates the possibility of many accidents to occur and also decreases the probability of occurrence and minimizes the consequences of the remaining ones. Fig. 1 shows the actual IRIS configuration. An IRIS first is the primary coolant pumps that are completely contained inside the vessel, thus eliminating any large vessel penetration and the possibility of secondary LOCAs.

The helical steam generators operate with superheated steam at the pressure of 5.8 MPa. The steam generators tubes operate in compression, since the primary water flows outside the tubes, thus steam generator tube rupture are less probable and its consequences are not as serious as in conventional SGs. Even in this case, the SGs are designed to withstand the primary system design pressure, and IRIS can operate with some SGs isolated in the case of tube failures.

The relatively high primary water mass, the low primary pressure drop and the high steam generators elevation enables enough natural circulation reducing the consequences of loss of flow accidents. The use of eight spool pumps inside the reactor vessel makes it possible to have a single locked rotor or pump seizure accidents without any core damage, moreover no reactor trip is needed from a safety point of view and the reactor could continue to operate at full power.

The IRIS pressure vessel is located inside a small spherical containment designed to a relatively high pressure. In the case of small or medium LOCA, the core remains covered for several days, or even weeks, depending on the natural heat removal over the containment external surface, without any emergency water injection. IRIS containment design makes use of suppression pools that are used also as a source of water gravity feed. For all analyzed accidents, the core remained covered. Further analyses are being made to show that IRIS can meet the requirement of "no need for off-site emergency response."

2.2. Optimized maintenance

IRIS concept considers a long life core: *refuelling intervals are four years*. To take full advantage of this core life, in terms of capacity factor, this should be matched by a similar maintenance interval. Following a previous work by MIT and its application to IRIS conditions, only seven items were identified as still needing resolution to attain the 48 months interval and their resolution is currently being addressed.

IRIS designers plan to make use of all recent advances in on-line monitoring and diagnosis plus in-service inspection. It is planned to develop specific techniques for all critical reactor components. Actual tasks involve bi-lateral research projects proposed within the International Nuclear Energy Research Initiative of the DOE (I-NERI Program) and within the Brazilian Energy Fund of the Science and Technology Ministry. Artificial intelligence based systems are being studied to provide additional operational support [9].

The IRIS project also includes some "design for maintainability" requirement envisioning time reduction in the main maintenance activity. As an example, the replacement of a steam generator needs only the opening of the pressure vessel and the uncoupling of few lines and devices: it is an easy task that can be accomplished in a short time schedule.

2.3. Long core life

An important approach to enhance proliferation resistance is to make the fuel less accessible by designing a core capable of operating in a straight burn long-cycle mode. IRIS considers a long life core: refuelling intervals can be as long as four years apart. Straight burn or partial refuelling shuffling cycles can be employed at the utility preference. This paper does not intend to present core design details, which can be found in references [10-11], which also presents links to more detailed references.

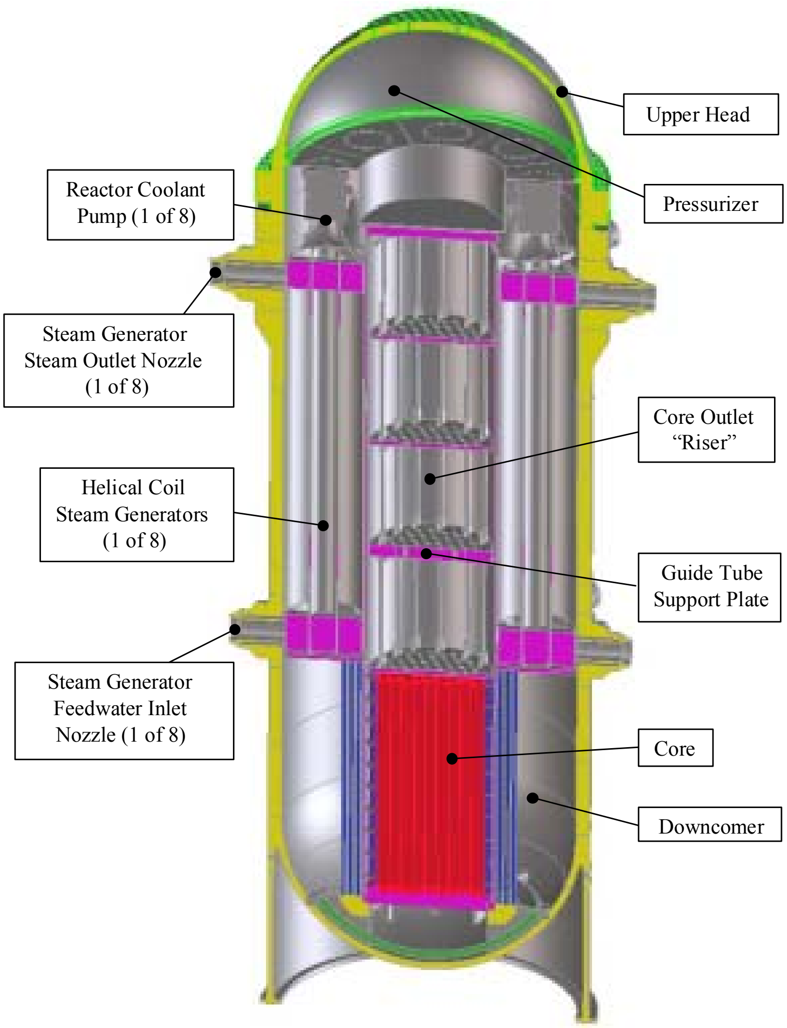

FIG. 1. IRIS integrated primary systems.

3. CNEN ACTIVITIES IN THE IRIS PROJECT

The main CNEN responsibility within the IRIS program is the internal pressurizer design and analyses. The design tasks for the pressurizer have included steam-water volume sizing, pressurizer to vessel physical separation by an internal thermal insulation, surge connections dimensioning and transient analyses, under different operating modes.

Since the signature of the collective agreement, the main tasks performed by the CNEN team have been:

— Conceptual design of the pressurizer, which includes the simplified transient analyses to evaluate the effectiveness of a steam bubble pressurizer, the conceptual design of the physical separation between the pressurizer saturated water and the reactor circulating water, the establishment of the safety valves set points, evaluation of heat losses from the pressurizer, specification of the heaters banks configuration, dimensioning of surge holes and re-circulation orifices of the pressurizer, study and definition of operational water levels;

— Development of a new RELAP 5 nodalization of the pressurizer;
— Development of simplified numerical tools to analyze operational transients, under different control options; and,
— Design experimental set-ups for the pressurizer study.

3.1. IRIS internal pressurizer design

Pressurizer type IRIS is not a self-pressurized system like Otto Hahn [12] or CAREM [13]. The high elevation-pumps have NPSH requirements that preclude the possibility of having a saturated water layer over the core, which characterizes a self-pressurized reactor. Previous studies have demonstrated that the best choice in such case is a pressurizer design similar to conventional PWRs, a water-steam system, with the vapour formation accomplished by electric heaters. This was the system proposed for the Safe Integral Reactor (SIR) [14].

Pressurizer requirements The pressurizer satisfies a dual function; controlling the system pressure and providing the water interface needed to the control of the coolant inventory. The basis for sizing the pressurizer also establishes the boundaries and requirements for the pressurizer pressure control system and for the pressurizer water level control. It is sized to meet the following requirements:

— The combined saturated water and steam volumes must be sufficient to provide the desired pressure response to system volume changes
— The water volume must be sufficient to prevent a reactor trip during a step-load increase of plus or minus 10% of full power, with automatic reactor control
— The water volume must be sufficient to prevent the uncovering of the heaters following a reactor trip and turbine trip, with normal operation of the control system and no failures in the nuclear steam supply system
— The steam volume must be large enough to accommodate the surge resulting from a step load reduction from 100% to house load without reactor trip, assuming normal operation of the control system
— The steam volume must be large enough to prevent water relief through the safety valves following a complete loss of load with the high water level initiating a trip and with no steam dump available
— A low pressurizer pressure S-Signal shall not occur because of a reactor trip and turbine trip, assuming normal operation of control and makeup systems and no failures of the nuclear steam supply system

Preliminary studies by CNEN indicated that a conventional water-steam system could meet the pressurizing system requirements, without the need of costly R&D activities. These studies showed that, with a proper design, the relation of Pressurizer Volume to Reactor Power could be 3.4 times greater than a conventional 2-loop Westinghouse plant and almost 2.7 times greater than the AP600 advanced reactor. This gave enough margin to avoid the need of a spray system, which really would represent a challenging effort for the engineering team. Simplified transient analyses showed that the IRIS pressurizer basic design configuration without spray can withstand a high pressure-dimensioning transient of 100% load rejection, without reaching the design pressure, assuming shut down only on the high pressurizer pressure [15].

Pressurizer housing The IRIS pressurizer is within the reactor head, which, as a part of the pressure retaining wall of the reactor pressure vessel, is designed as a Class 1 vessel according to the ASME Code Section III. All penetrations of the closure head are located on the upper dome. There are 45 nozzles for control rod drive mechanisms (CRDMs), 90 nozzles for heating rods, and 48 instrumentation nozzles. The nozzles for the CRDMs and for the instrumentation tubes which extend through the head region have internal pipe extensions that go through the bottom plate.

Figure 2 shows the pressurizer and how the saturated water is separated from the reactor circulating sub-cooled water by an internal structure with an “inverted hat shape”. The function of this structure includes: (a) preventing the head closure flange and its seals from being exposed to the temperature difference between the reactor and pressurizer water, reducing thermal stresses and maintaining the sealing tightness; (b) providing some thermal insulation to reduce the heat transfer and maintain an adequate saturated water layer within the pressurizer; (c) providing structural support for the control rod drives mechanisms (CRDM) guides, core instrumentation tubes, and heaters; and (d) providing the communication flow paths between the reactor and pressurizer for surge and re-circulation flows.

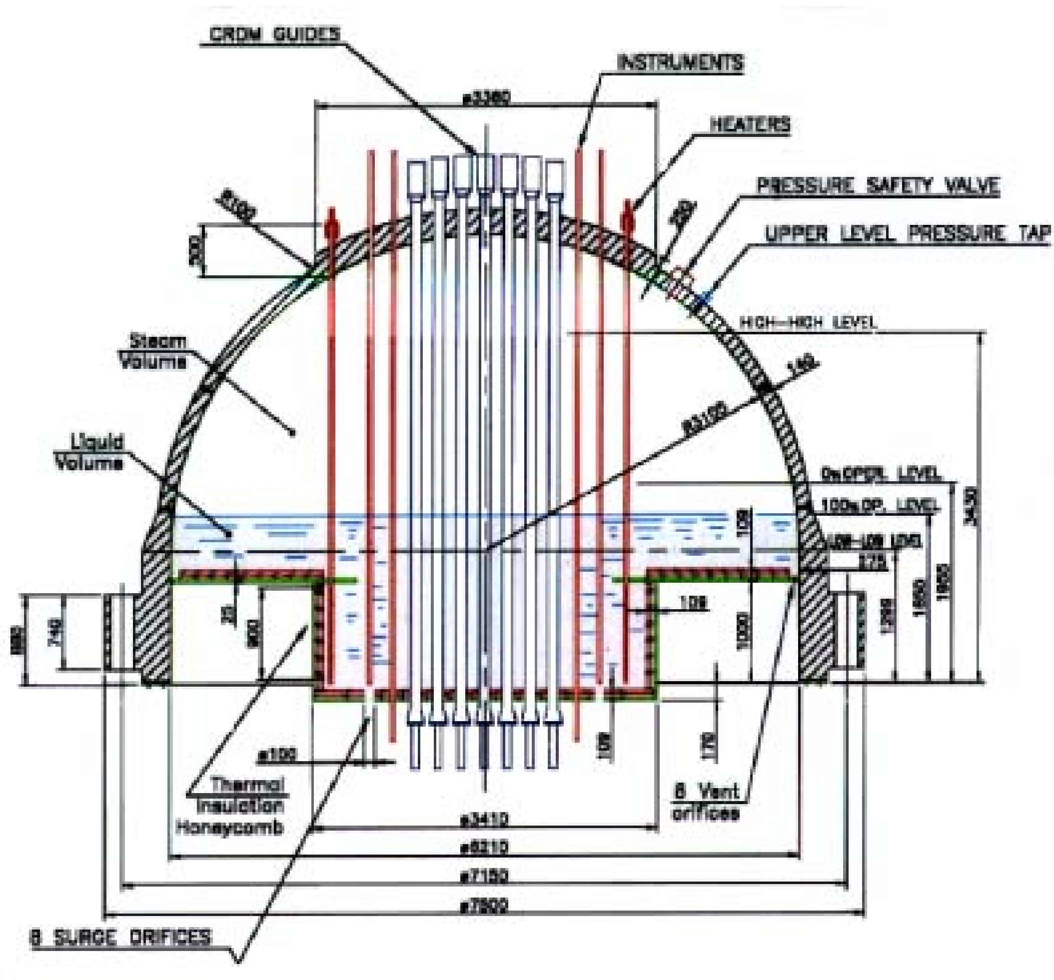

FIG. 2. IRIS integral pressurizer.

Spray Analyses performed indicate that, after an in-surge transient, the return to pure steady state conditions is more sluggish than if a spray system were present. Simplified analyses of consecutive power steps had shown that this small unavoidable draw back of the current IRIS design does not affect the capability of the pressurizer to adequately cope, even when an out-surge operational transient happens after an in-surge transient. Thus, a conventional pressurizer spray capability is not necessary in the IRIS, however, special passive spray design concepts might be considered in future efforts to further improve the pressurizer performance.

Heaters The pressurizer heaters are designed to create and maintain the saturated water layer and to produce enough steam to limit the pressure decrease during transients of power increase. There are 90 electrical heaters providing a total heating power of 2430 kW. They are grouped in two banks, a proportional and a backup bank. A PI controller is proposed to control the proportional heater. The use of this controller can improve the pressurizer pressure response mainly during slow transients.

3.2. IRIS Internal Pressurizer Analyses

The requirements for the pressurizer sizing define a set of transients to be analyzed. Table I lists these IRIS design operating transients. This paper presents few results for one of them.

Transients that result in large and fast pressurizer in-surge were simulated and reported previously [15] to demonstrate the viability of the current pressurizer design without spray. Simplified adiabatic models and also more realistic RELAP5 mod3.3 models showed satisfactory results, allowing to go ahead with the present design solution. The ability to cope with very large in-surge transients was tested using some very demanding cases, e. g. full load rejection, with a 1s delayed shut down – after the occurrence of the high pressure signal – and a 4s delayed actuation of the Passive Emergency Heat Removal System (PEHRS) for decay heat removal. The pressure response in all cases remained within limits and, even with no credit for any power operated relief valves actuation, the safety valve pressure set point was not reached.

Table I. IRIS Design Operating Transients

Step Load changes of plus or minus 10% of full power
Ramp load increases and decreases of 5%/minute
Daily load follow operations: power ramp from 100 to 50% in 2 hours; operation at 50% power for 2 to 10 hours; power ramp from 50 to 100% in 2 hours; power remains at 100% for the remainder of a 24-hour cycle
Grid frequency response resulting in a maximum of 10% power changes at 2% per minute.
Full load rejection following a turbine trip

Although the thermal-hydraulic code chosen for the IRIS accident and transient analyses is the RELAP5 mod3.3, the model prepared for accident analyses does not have the normal operation controls needed for operational transients analyses. A simplified tool was developed for this purpose [16]. It was implemented using a set of interrelated MS Excel and Visual Basic macro programmed worksheets. It allows a quick review of the pressurizer data and execution of simplified transient analyses of its behaviour under different temperature/power programs, reactor and control characteristics, pressurizer level programs and volumetric control system capacities. Reactor kinetics were not modelled in the initial phase of the development of this tool, the transients are applied to the thermal power at the secondary side of the steam generators and the control were assumed to act direct on the reactor power, controlled within allowable rates of power change.

The pressure, level and reactor power controls used are similar to that of most conventional PWR reactors. The results presented here were obtained considering a partitioned sliding averaged temperature program (Fig. 3) and a pressurizer constant level program. Fig. 4 shows results for the initial part of the 10% negative step load transient, in terms of the steam generator (QSG) and the core thermal power (QR).

The control parameters used produced a fast and stable power response. The temperature results are in Fig. 5, where T_C is the mean core temperature, T_R is the riser temperature, T_{SG} is the mean steam generators temperature, T_D is the mean downcomer temperature. The behaviour of the pressurizer water level is shown in Fig. 6 and the pressure in Fig. 7. The pressure and temperature responses were fast and with a stable convergence to the programmed values without overshoot. The results showed excellent pressure smoothing capability of the pressurizer.

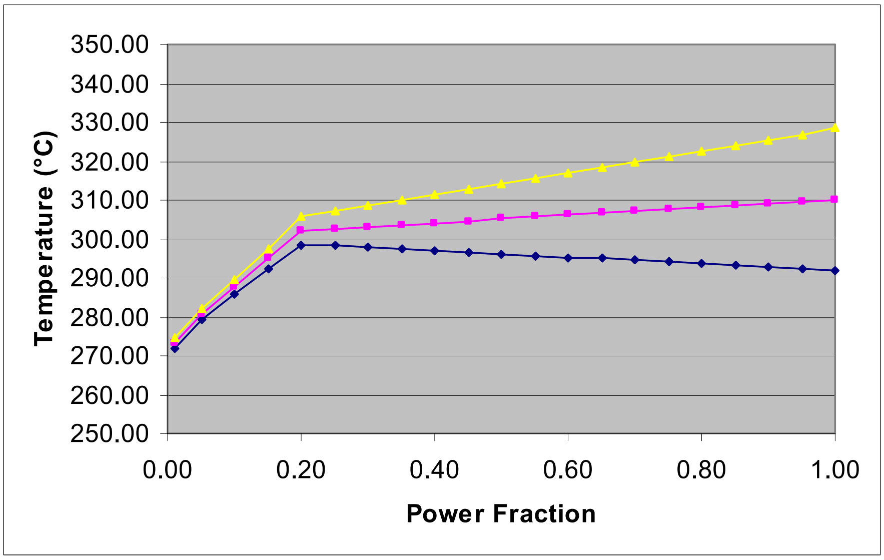

FIG. 3. Sliding T average program.

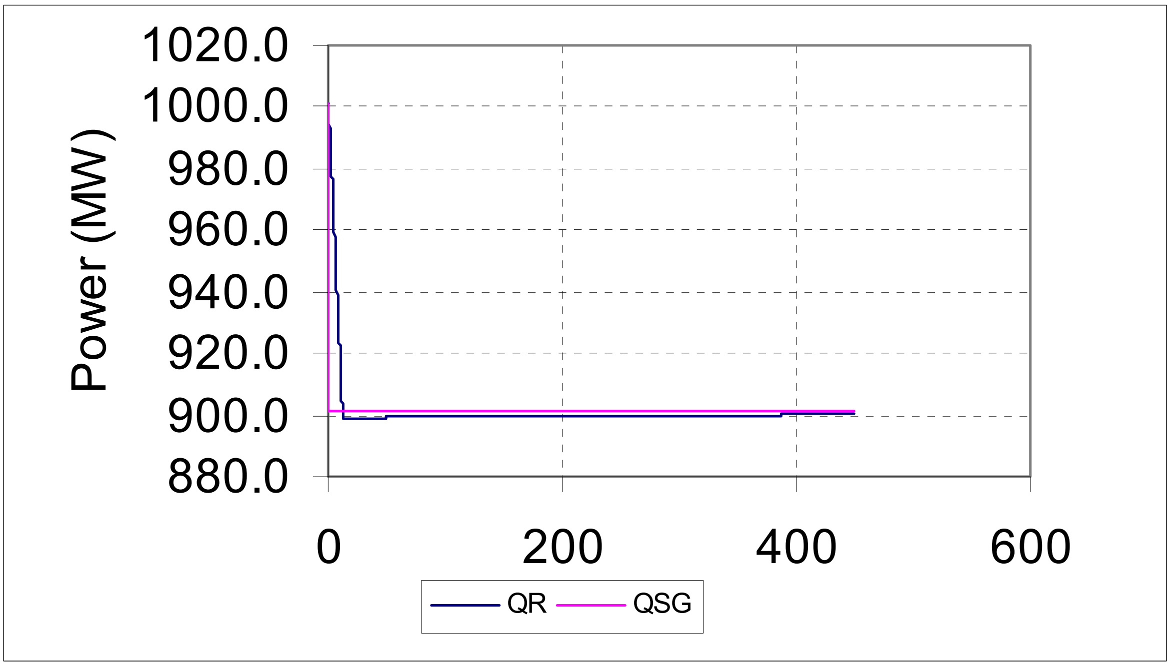

FIG. 4. Step load of –10%: Power response.

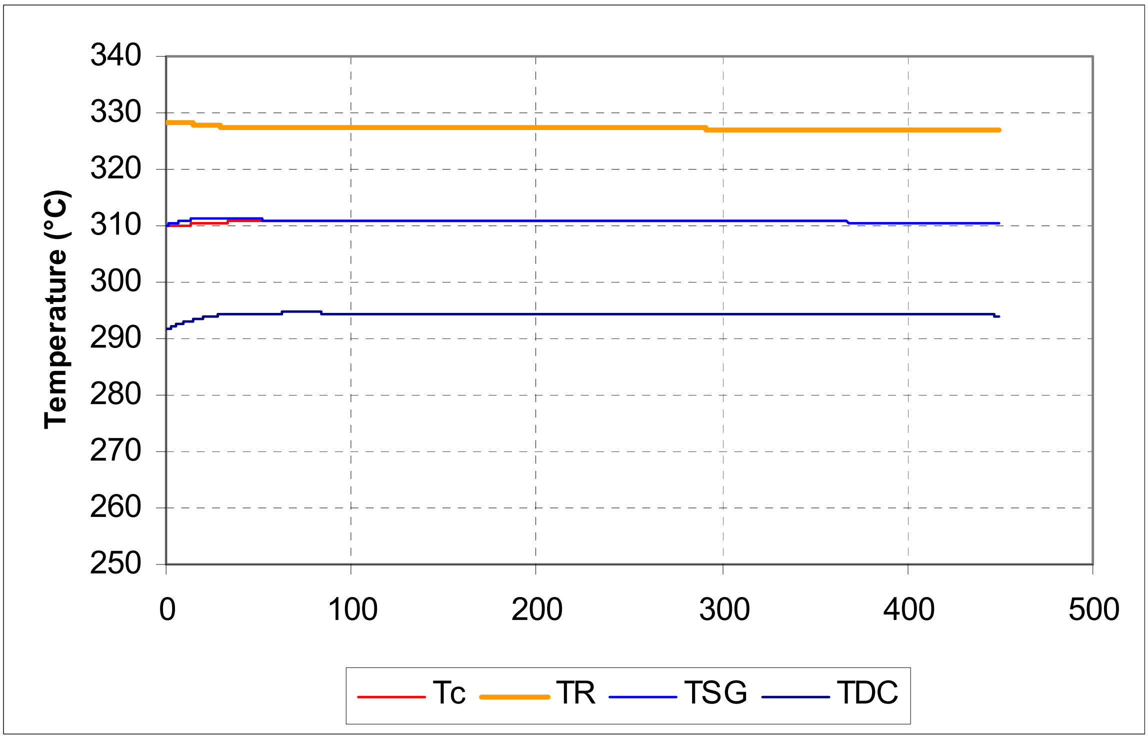

FIG. 5. Step load of –10%: Temperature.

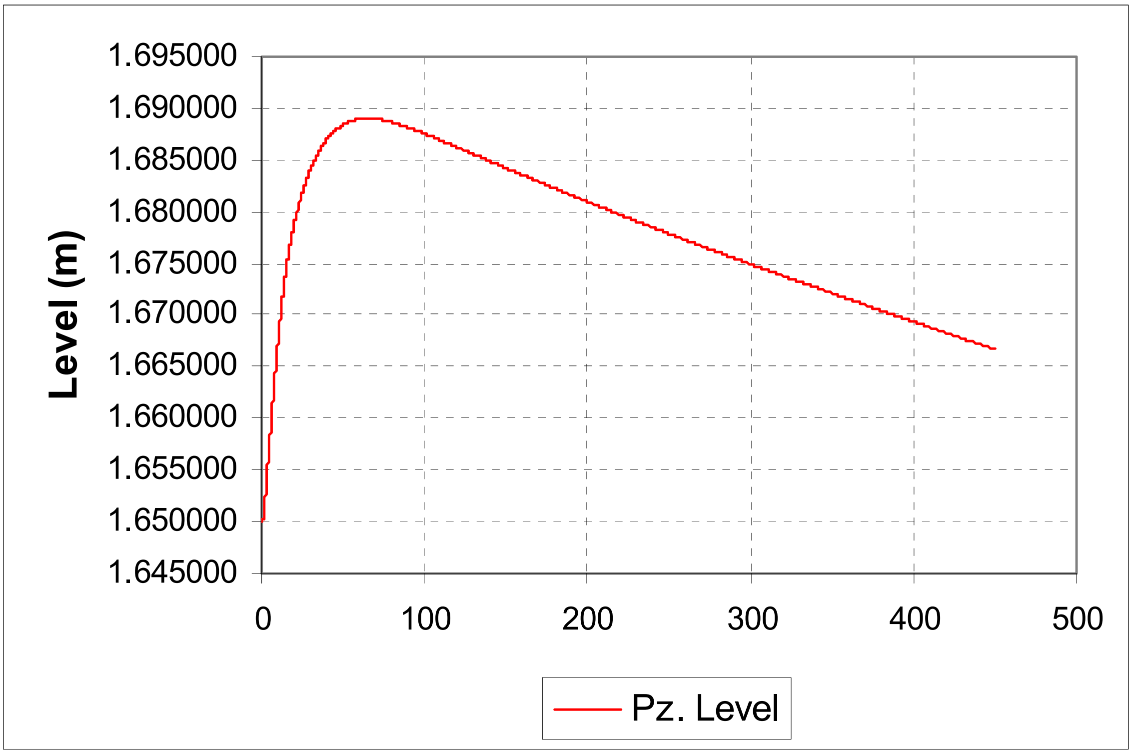

FIG. 6. Step load of –10%: Level.

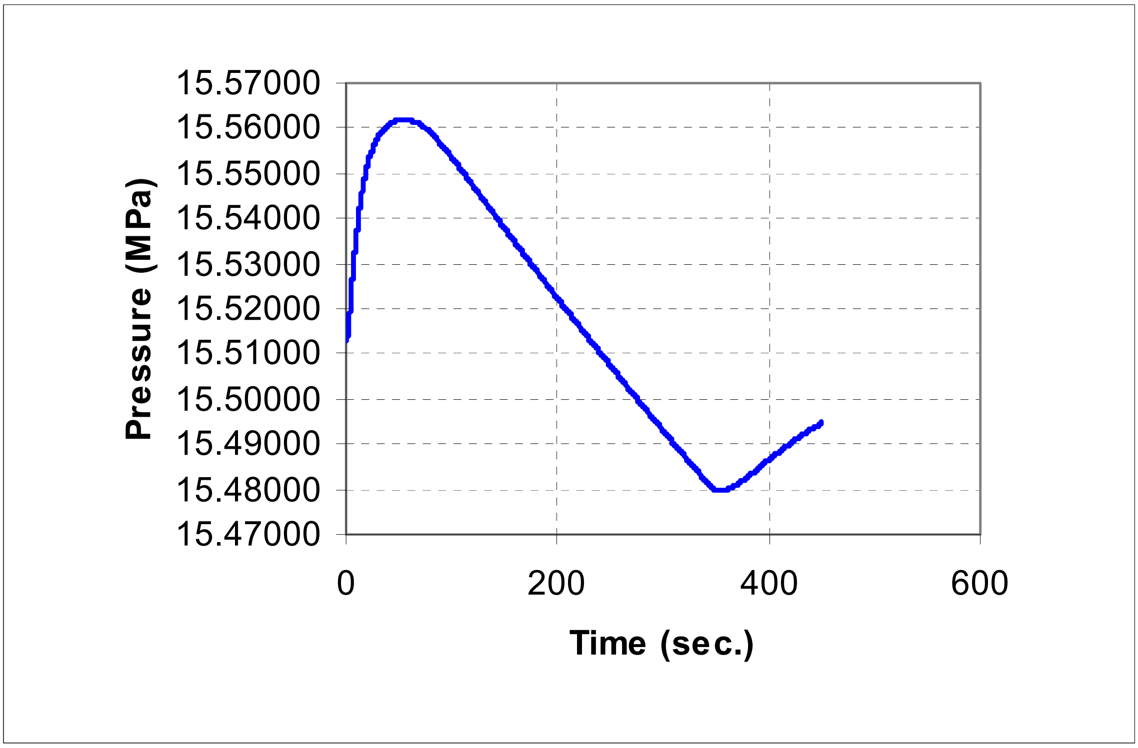

FIG. 7. Step load of –10%: Pressure.

4. IRIS CONSORTIUM MANAGEMENT ISSUES

The self-sustained activity of the IRIS consortium partners, has greatly exceeded the work scope originally scheduled in response to the DOE NERI request for proposal. Essentially, the difference resulted to be between conducting an exploratory investigation of a concept feasibility versus initiating a real design focussed towards commercial feasibility. Items like component characterization or site layouts were never considered in the proposal but they are addressed here at length.

Managing a consortium comprising 22 organizations from nine countries, which vary from industry to academia, to research laboratories and power producers, and span 7 languages and 17 time zones is a challenge to say the least. However the IRIS Consortium is held together and thrives because:

- All members share the belief that the IRIS concept can contribute significantly to the forthcoming nuclear renaissance and are firmly committed to see it happening; and,
- The unique IRIS concept where all members of the consortium, regardless of the level of their individual contributions, are “owners” of IRIS and share all information and decisions.

The diversity in culture and technical background and objectives is turned into a positive thrust by providing different viewpoints through the operation of multi-organization teams. In the case of CNEN for example, CNEN leads and integrates the contributions of Westinghouse and Oak Ridge National Laboratory to the pressurizer design. For the RELAP analyses, CNEN provides an independent verification to the input file preparation by the University of Zagreb, Croatia. This represents a necessary contribution to the verification and validation of one of the major computer codes used in the IRIS safety analyses. In addition to the operation of multiple working groups, the entire IRIS team meets twice a year, once in the US and once at one of the overseas members, to review work in progress, outline future work and discuss/resolve technical and programmatic issues.

Working on a new program being developed at a brisk pace by an international team is an extremely exciting endeavour enjoyed by all team members, but it carries a few practical drawbacks. A part from the quite heavy coordination burden on the program manager, it resulted in the following “glitches” which must be brought to the attention:

- With the work proceeding in parallel at various establishments, the length of the various tasks being different and the design philosophy and design characteristics (core fuel lifetime, power rating, configuration, etc.) changing, it happened that at times different members of the consortium were working on different design versions.
- Quite an effort has been exercised in trying to homogenize the writing style of so many different contributions and mother tongues.
- Preparing and issuing annual reports takes an enormous amount of time. We believe that it was well worth to document to the best of our abilities all the work performed by the IRIS consortium.

The coordination of each national team presents similar problems. The Brazilian Team coordination was not an easy task. Regional differences, the concurrence of several different tasks, the integration of people from research institutes and factory and also the limited funds for such activities are the main difficulties. The key to keep a highly motivated team sets in the maintenance of a high level of research demand accompanied by frequent meetings and the offer of international knowledge exchange. The need for the edition of two annual reports to be presented during the two annual international meetings is almost enough to establish the high level in research demand.

During the time that elapsed from the agreement signature, it was observed that such kind of project allows the improvement of the Brazilian nuclear knowledge, once the research and engineering teams are kept involved with activities shared with the most advanced teams from developed countries. We feel that this project has the capability of expanding the present level of deployment of nuclear technology in Brazil.

5. CONCLUSIONS

This paper presented a very brief description of IRIS, with emphasis in its “safety by design approach.” A summary of the Brazilian activities in the IRIS project, coordinated by CNEN, was presented. Some of the management issues concerning this large and geographically dispersed design work force was also addressed, demonstrating that such approach of international project immediately found a positive resonance, as the IRIS team kept growing over its three-year life from the initial four members and two countries to the present over 22 members from nine countries.

The IRIS project represents the latest evaluation of the light water reactors, which are by far the predominant type of nuclear reactors world-wide. Its technical, economic and programmatic characteristics, position IRIS to be a very significant contributor to nuclear power production in the 21st century. Its innovative approach to international, co-operative management enables IRIS to be deployable world-wide. Brazil recognizes that IRIS can be an important component of its nuclear portfolio and is contributing to IRIS development, including a leading role in the pressurizer design and analysis.

The pressurizer is a significant component of the IRIS safety by design approach by intrinsically smoothing the response to overpressure transients and of its design simplification, by making sprays unnecessary. Analyses performed by CNEN and summarized here confirm the robustness of the IRIS pressurizer design.

6. REFERENCES

[1] Department of Energy, Office of Nuclear Energy, Science and Technology, “Generation IV - Nuclear Energy Systems, http://gen-iv.ne.doe.gov/, 2002.

[2] CARELLI, M. D., et al., “IRIS, International New Generation Reactor,” Proceedings of the 8th International Conference on Nuclear Engineering, Baltimore, USA, April 1-5, 2000.

[3] CARELLI, M.D., et al, “IRIS Reactor Development,” Proceedings of the 9th International Conference on Nuclear Engineering, Nice, France, April 8-12, 2001.

[4] CARELLI, M. D., et al, “Trade-off Studies for Defining the Characteristics of the IRIS Reactor Core,” 16th International Conference on Structural Mechanics in Reactor Technology, Washington, USA, August 12-17, 2001.

[5] PETROVIC, B, et al, “International Collaboration and Neutronic Analyses in Support of the IRIS Project,” Trans. Am. Nucl. Soc. 83, pp. 186-187 (2000).

[6] ORIANI, L., et al, “Thermal Hydraulic Tradeoffs in the Design of the IRIS Primary Circuit,” Proceedings of the 9th International Conference on Nuclear Engineering, Nice, France, April 8-12, 2001.

[7] CONWAY, L.E., et al, “Simplified Safety and Containment Systems for the IRIS Reactor,” Proceedings of the 9th International Conference on Nuclear Engineering, Nice, France, April 8-12, 2001.

[8] CARELLI, M.D., et al, “SAFETY BY DESIGN: A NEW APPROACH TO ACCIDENT MANAGEMENT IN THE IRIS REACTOR,” IAEA-SR-218-36, IAEA

INTL. SEMINAR ON STATUS AND PROSPECTS OF SMALL AND MEDIUM SIZED REACTORS, CAIRO, EGYPT, MAY 27-31, 2001.

[9] BAPTISTA F., B.D., BARROSO, A.C.O., "IDENTIFICATION OF IRIS REACTOR TRANSIENTS WITH SELF-ORGANIZED MAPS ", Paper 1185, approved for the GENES4/ANP2003 Conference, Sep. 15-19, 2003, Kyoto, JAPAN.

[10] PETROVIC B., et al. “FIRST CORE AND REFUELINOPTIONS FOR IRIS,” ICONE10-22581, Track 6, in Proceedings of ICONE10, 10th International Conference on Nuclear Engineering, Arlington, VA, April 14-18, 2002.

[11] PETROVIC, B., et al, “Innovative Features and Fuel Design Approach in the IRIS Reactor,” Workshop on Advanced Reactors With Innovative Fuel (ARWIF-2001), Chester, U.K., October 22-24, 2001.

[12] Nuclear Merchant Ship “Otto Hahn.” Safety Assessment GKSS (1968).

[13] DELMASTRO, D., et al, “CAREM: AN ADVANCED INTEGRATED PWR,” International Seminar on Status and Prospects for Small and Medium Sized Reactors, Cairo, Egypt, 27-31 May 2001.

[14] MATZIE, R. A., LONGO, J.,. BRADBURY, R.B, “Design of the Safe Integral Reactor”, Nuclear Engineering and Design, Vol. 136, Issues 1-2, Pages 73-83, 1992.

[15] BARROSO, A.C.O., BAPTISTA F., B.D., PALMIERI, E.T., SABUNDJIAN. G., ANDRADE, D.A., MACEDO, L.A., "CNEN IN THE IRIS PROJECT", LAS-ANS Symposium 2002 – Power Supply and its Challenges: The Nuclear Proposal, 17-20 June, 2002, Rio de Janeiro, Brazil.

[16] BARROSO, A.C.O., BAPTISTA F., B.D., ARONE, I.D., MACEDO, L.A., SAMPAIO, P.A.B., MORAES, M., “IRIS PRESSURIZER DESIGN”, Proceedings of ICAPP ’03, Paper 3227, Córdoba, Spain, May 4-7, 2003.

INNOVATIVE FUEL CYCLES

(Session 6.2)

Chairpersons

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INDIA

INTEGRATED EXPERIMENTS TO DEMONSTRATE INNOVATIVE REPROCESSING OF METAL AND OXIDE FUEL BY MEANS OF ELECTROMETALLURGICAL TECHNOLOGY

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Abstract. Electrometallurgical pyro-processing is a key innovative technology to realize closed actinides cycle with keeping high proliferation-resistance and economy. The current status and the strategy of development of the electrometallurgical pyro-processing technology are summarized, and the necessity to demonstrate whole process in one continuous operation is stressed. The test titled "integrated experiments of electrometallurgical reprocessing of metal and oxide Pu-containing fuel" has been jointly started by CRIEPI and JNC. The new glove box system equipped with pyro-process apparatuses such as electrorefiner was developed and installed in CPF of JNC-Tokai Works. The test will enable us to evaluate the technical feasibility of this process to apply closing actinides cycle.

1. INTRODUCTION

Dry (i.e., non-aqueous) processing technologies are currently being focused in many countries for closing actinide fuel cycle because of their favorable economic potential [1] and an intrinsic proliferation-resistant features due to inherent difficulty of extracting weapons-usable plutonium [2]. Electrometallurgical technology (pyro-process) originally developed for metal fuel by Argonne National Laboratory (ANL) [3] is one of the most attractive dry processing technologies, because it has an inseparability properties of Pu from other actinides in any step of the process [4]. This property enables us to enhance intrinsic proliferation resistance in addition to recovery of long-lived transuranium elements for transmutation in the reactor without addition of further treatment. Intensive fundamental studies in many organizations such as CRIEPI[5], ITU[6], BNFL&CEA[7], etc. as well as the development to engineering scale in ANL such as the test to treat spent EBR-II fuel[8] have demonstrated high potentiality to industrialize.

Inseparability of plutonium from other actinides in the refining step of pyro-process is theoretically guaranteed by the thermodynamic properties such as the distribution of actinides in the molten salt and liquid cadmium system expressed as follows,

$$SF_M(\mathrm{Pu}) = \frac{[X_M]_{salt}}{[X_M]_{cd}} / \frac{[X_{Pu}]_{salt}}{[X_{Pu}]_{cd}}$$

$$= \frac{[\gamma_{Pu}]_{salt}}{[\gamma_{Pu}]_{cd}} \frac{[\gamma_M]_{Cd}}{[\gamma_M]_{salt}} * \exp\left[\frac{\Delta G_f^o{}_{MCl3} - \Delta G_f^o{}_{PuCl3}}{RT}\right], \qquad (1)$$

where $[X_i]_j$ denotes the concentration of i in phase j. Notation of this equation follows the ordinal expression of thermodynamics. The distribution data set of the actinides, U, Np, Pu, Am, and Cm, was first reported by one of the authors in 1992 [9] as Table I. It shows clearly the recovery of plutonium shall take place simultaneously with the recovery of other actinides with keeping the separation of fission products such as lanthanides.

Applicability of the electrometallurgical technology to treat other fuel types such as oxide fuel, nitride fuel, MOX fuel, etc. is another attractive feature. Conversion technology such as Li reduction [10] enables to feed oxide fuel into electrometallurgical pyro-process. Figure 1 shows the example of reduced metal from simulated spent MOX fuel with Li reductant carried out by one of the authors [11]. Nitride fuels are found to be processed in the electrometallurgical technology without major reformation of the process [12].

The molten salt media also gives two important advantageous properties as a solvent material in nuclear processing. The radiation stability of molten salt allows the processing of spent fuels of high radioactivity (e.g. spent fuel with short cooling time) without any increase of the solvent waste. Since molten salt is not a neutron moderator such as water is, comparatively large amount of fissile material can be handled in the process equipment, i.e. experimental facilities are compact and economical.

The major wastes specific to pyro-process are the salt waste and the metal waste. The salt waste must be immobilized into water-insoluble form. Several immobilization methods have been proposed[13,14], and immobilization in the glass bonded sodalite form was demonstrated with irradiated fuel treatment[15]. Consistency of the metal waste form has also been studied with simulated waste[16].

Table I. Measured separation factors (SF) of actinides from Pu in LiCl-KCl/Cd system [9]

Element	Separation Factor
U	0.532
Np	1.13
Pu	1.00 (basis)
Am	1.64
Cm	1.87
Ce	24
Nd	21

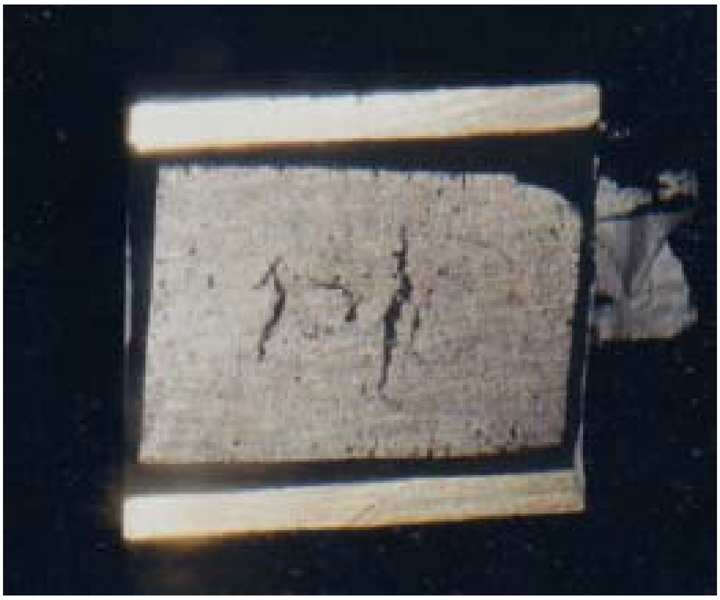


FIG. 1. Reduced metal from MOX [11].

2. CLOSING ACTINIDES CYCLE WITH PYRO-PROCESSING TECHNOLOGY

Central Research Institute of Electric Power Industry (CRIEPI) has been proposing the "actinide recycling by pyro-process with fast breeder reactor" [5], where metal fueled FBR is combined with pyrometallurgical reprocessing, pyrochemical reduction of oxide fuel into metal form, and pyro-partitioning of TRUs from HLW of PUREX reprocessing of LWR fuel. The concept is schematically described in Fig. 2, where the processes with pyro-processing technology are noted with red letters. As seen in this figure, pyro-processing technology can be applied to fill up the gap between two cycles as well as to close FBR fuel cycle itself. The detail of each application is summarized in this section with the technical basis developed by CRIEPI

CRIEPI started the study on actinide recycling technology on 1985 including the joining in IFR Project of US-DOE from 1989 to 1995. \Pyro-partitioning of TRU and transmutation [17] was nominated to the OMEGA project (long-term research/development program for the long-lived nuclides partitioning and transmutation) of Japan. The project shared by Japan Atomic Energy Research Institute(JAERI), Japan Nuclear Cycle Development Institute(JNC) and CRIEPI was checked and reviewed on 1999 by Japanese Atomic Energy Commission. On 1998, CRIEPI and JRC-ITU started joint study to demonstrate the feasibility of each actinide recycling process with using real spent fuel and real HLW [18].

In the *FBR fuel cycle* of Figure 2, metal fuel consists of U-Pu-Zr and U-Zr alloys are fabricated by the injection casting process that allows remote handling and high MA content. Technical feasibility of 20 kg U-10Zr batch fuel casting[19] and the allowable amount of MA in U-Pu-MA-RE-Zr[20] have been confirmed experimentally by CRIEPI. The irradiated metal fuel is reprocessed by pyro-reprocessing. The chopped spent fuel is anodically dissolved in LiCl-KCl molten salt *electrorefining process* (Fig. 3), and U-Pu and U are recovered to the liquid Cd cathode and the solid iron cathode, respectively. The U-Pu products of the process contain MAs and rare earth fission products according to the electropotentials shown in Fig. 4[21]. The recovery of U, Pu and Am from U-Pu(Am)-Zr fuel[22] as well as the basics of molten salt electrorefining[23,24] has been studied. The accompanied media, salt and Cd, are to be removed from U-Pu-MA or U products by distillation at the *cathode process* (Fig. 5). The obtained U-Pu-MA or U ingot is casted into quartz mold to make metal slugs at injection casting fuel fabrication. On the other hand, the fission products remained in the salt or metal phase are recovered and converted into the ceramic waste [14] or metal waste.

The oxide fuels such as UO_2 and MOX can be treated in the above mentioned FBR cycle after reduction to metal form as Figure 2. The *Li reduction process* using Li metal reductant has been intensively studied by CRIEPI[11,25,26]. As shown in Fig. 6, dissolved Li react with oxide to form Li_2O in the LiCl molten salt at 650°C. The feasibility of this process has already been demonstrated with UO_2 [25], MOX [11] and single MA oxide like AmO_2 [26].

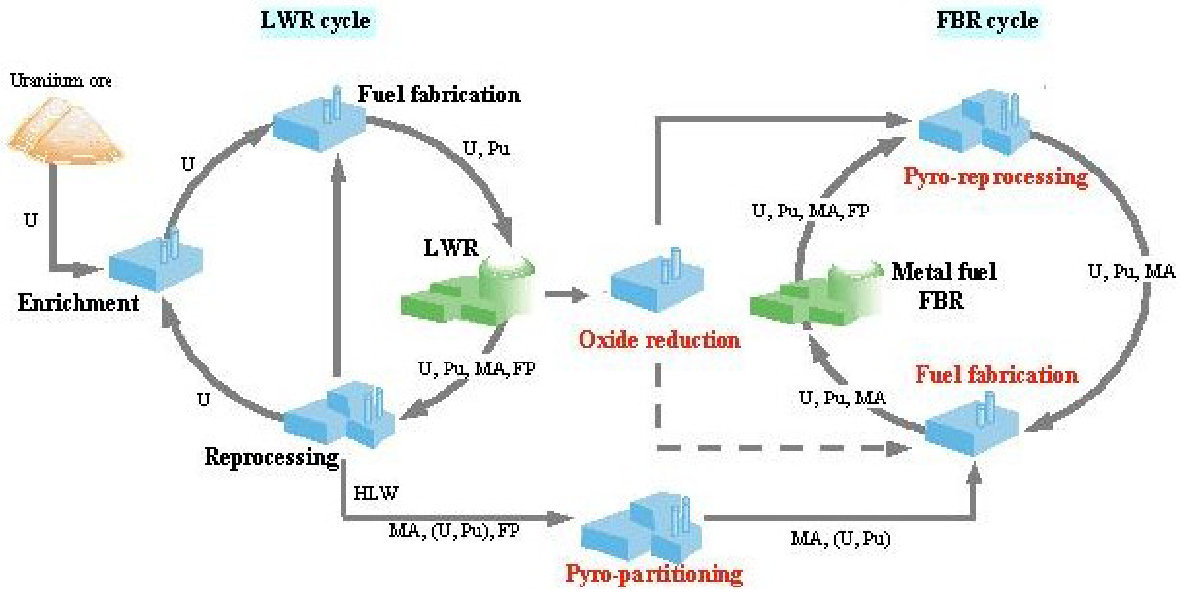


FIG. 2. CRIEPI's concept to close the actinides cycle by applying pyro-technology.

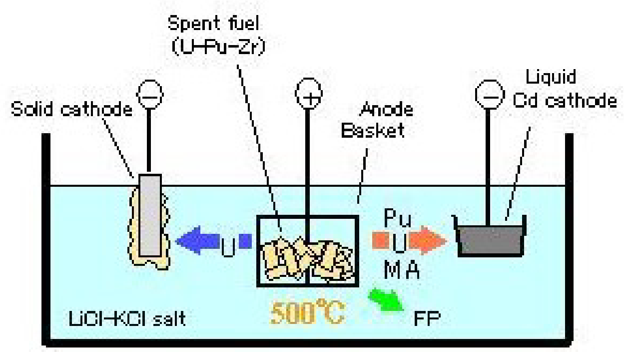


FIG. 3. Electrorefining process.

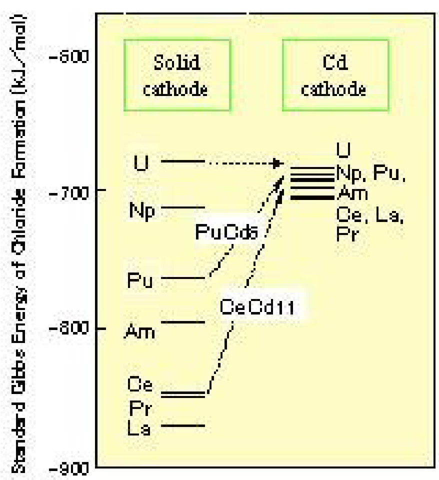
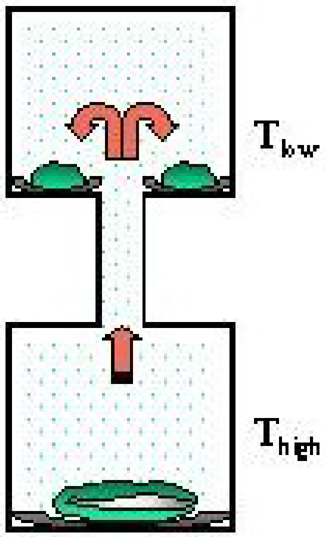


FIG. 4. Reduction potentials in LiCl-KCl.



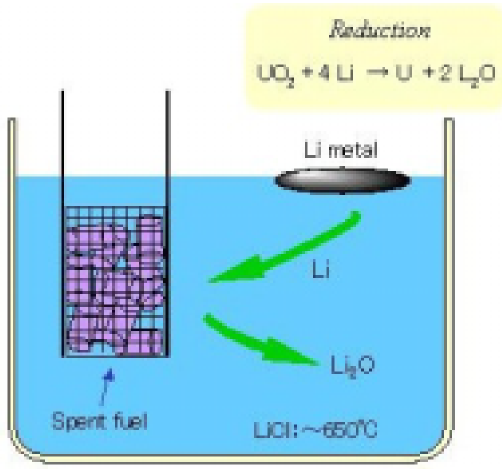


FIG. 5. Cathode processing. *FIG. 6. Li reduction of oxide fuel.*

The *pyro-partitioning* of TRUs from HLW of PUREX reprocessing of LWR fuel is also adopted as shown in Fig. 2. The HLW is converted into chloride through chlorination, and TRUs are extracted by molten salt extraction process followed by eletro-recovery. The feasibility of the pyro-partitioning has already been demonstrated with unirradiated TRUs [27, 28]. The test with real HLW is now under study [18].

3. FEASIBILITY STUDY OF FAST REACTOR CYCLE SYSTEM

At 1998, JNC and utilities agreed to start studying jointly to establish a commercialization concept which maximizes economic competitiveness while ensuring the highest levels of safety[29]. The study was named as "Feasibility study of commercialized fast reactor systems", and consists of 3 phases listed below.

— ***Phase 1 (1999-2000)*** Evaluate a wide variety of technical options introducing innovative technologies and select several useful concepts and prepare necessary R&D schedule.
— **Phase 2 (2001-2005)** A well-balanced consistency between FR system and its fuel cycle is pursued for FR cycle candidate concepts screened in Phase 1. Promising commercialized FR cycle candidates (2 or 3) are determined through design studies and engineering tests.
— ***Phase 3+(2006-2015)*** *Conduct the study with Check & Review every 5 years and present FR cycle technology with the technical data in around 2015*

In order to achieve the objective of each phase, this study is being carried out with the participation of all parties concerned in Japan, i.e., the electric utilities, CRIEPI, and JAERI.

Key technologies for advanced fuel cycle system have been proposed and reviewed. As the end of phase-1, several candidate technologies were screened and selected by preliminary assessment. As the advanced reprocessing candidates, dry processing such as the electrometallurgical process and the oxide electrowinning process have been selected in addition to the aqueous processes.

In the study of advanced fuel cycle technology, R&D of the electrometallurgical technology has been designated to CRIEPI. The technical achievement of CRIEPI are reviewed in the feasibility study. In the course of this study, the importance to carry out the electrometallurgical processing test with using plutonium was recognized. Because the recovery of transuranics has been examined only for each step [18] or U based fuels [8, 30], it is necessary to test whole process in one continuous operation with using Pu-containing fuel. Hence, CRIEPI and JNC have agreed to start new joint study to implement the integrated experiments of electrometallurgical reprocessing of metal and oxide Pu-containing fuel at *chemical processing facility* (CPF) of JNC-Tokai Works [31].

4. INTEGRATED EXPERIMENTS TO DEMONSTRATE REPROCESSING OF METAL AND OXIDE FUEL BY MEANS OF ELECTROMETALLURGICAL TECHNOLOGY

4.1. Test schedule and process to demonstrate

The electrometallurgical process selected for the integrated experiments is shown in Fig.7. It consist of (1) reduction of oxide fuel into metal form by means of Li reductant, (2) molten salt electrorefining to recover U and U-Pu, (3) cathode process to remove salt and Cd from actinides, (4) injection casting of actinide to form metal fuel, and (5) oxidation of actinide to form oxide fuel. As shown in the same figure, the experiments proceeds in the order of (2), (3), (4) for metal fuel reprocessing and (1), (2), (3), (5) for oxide fuel reprocessing, respectively.

The time schedule of this study is shown in Table II. In year 2002, the apparatus was finished installing, and the cold test with non-radioactive simulants started. The integrated test with uranium is planned on 2003, and followed by the test with plutonium. As shown in this figure, we are now in the phase of cold & uranium test. In the course of process experiments, development of auxiliary technology such as nuclear material management and waste management are carried out in order to evaluate the technology to realize electrometallurgical reprocessing.

4.2. Apparatus development

The new glove box system has been developed, and installed at CPF. It consists of one process glove box with an Ar purification unit and two air glove boxes(see Fig.8). The atmosphere of Ar glove box is kept in less than 10ppm of O_2 and H_2O impurities for avoiding the deterioration of molten salt and recovered metals. The radiation-shielded heating wells are placed on the bottom of the box to install process equipments such as electrorefiner. Up to 220g of Pu(+U^{235}) and $2.22*10^8$Bq of fission products can be handled in this box.

The process equipments are developed based on the study of CRIEPI (see chapter 2), and of JNC [32]. The installed Li reduction apparatus is shown in Fig. 9. Small tungsten crucible charged with oxide is immersed in the LiCl bath contained in the larger crucible. About 20 g of MOX will be reduced in this apparatus. The installed electrorefiner is shown in Fig. 10. It consists of a steel crucible of 120 mm diameter for salt electrolyte and three electrodes, an anode, a cathode and a reference electrode. Two different cathodes, a solid cathode and a liquid Cd cathode, are designed to use each other. The anode assembly is designed to charge the reduction crucible for anodic dissolution of the reduced oxide. The installed cathode processor is shown in Fig. 11. The deposit from the electrorefiner is charged in a graphite crucible to be induction heated, and the vaporized salt and Cd is recovered at the condenser-collector located above the graphite crucible. The recovered U-Pu or U from cathode processor is charged into another graphite crucible for same induction heating, and the melt is injection casted into the quarts mold for obtaining slug samples. The oxidation of metal with Ar-O_2 gas is carried out in the apparatus shown in Fig. 12.

In year 2002, the installation of apparatus was completed, and the internal safety review to start active experiments with U and Pu is underway.

Table. II. Time schedule of the integrated experiments

4.2.1.1. JPN FY	*2000*	*2001*	*2002*	*2003*	*2004*	*2005*
Licensing, design & installation						
Process experiments			***Cold & U test***		***Pu test***	
Auxiliary technology development						

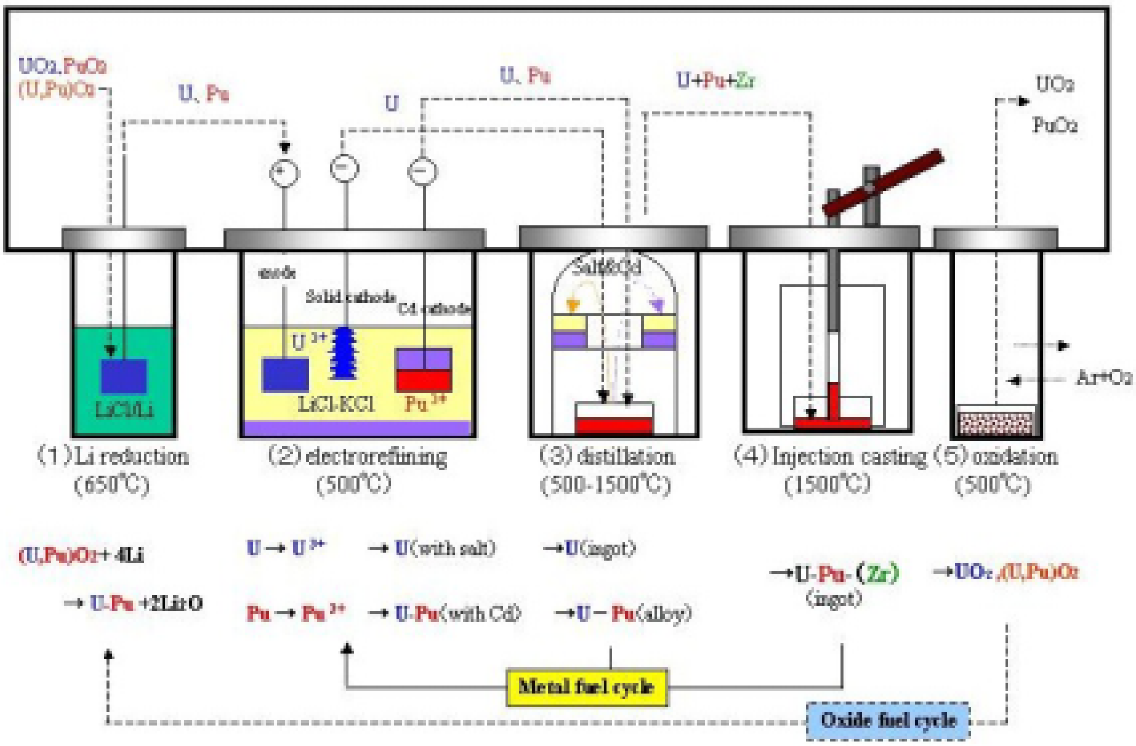


FIG. 7.Electrometallurgical process for the integrated experiments.

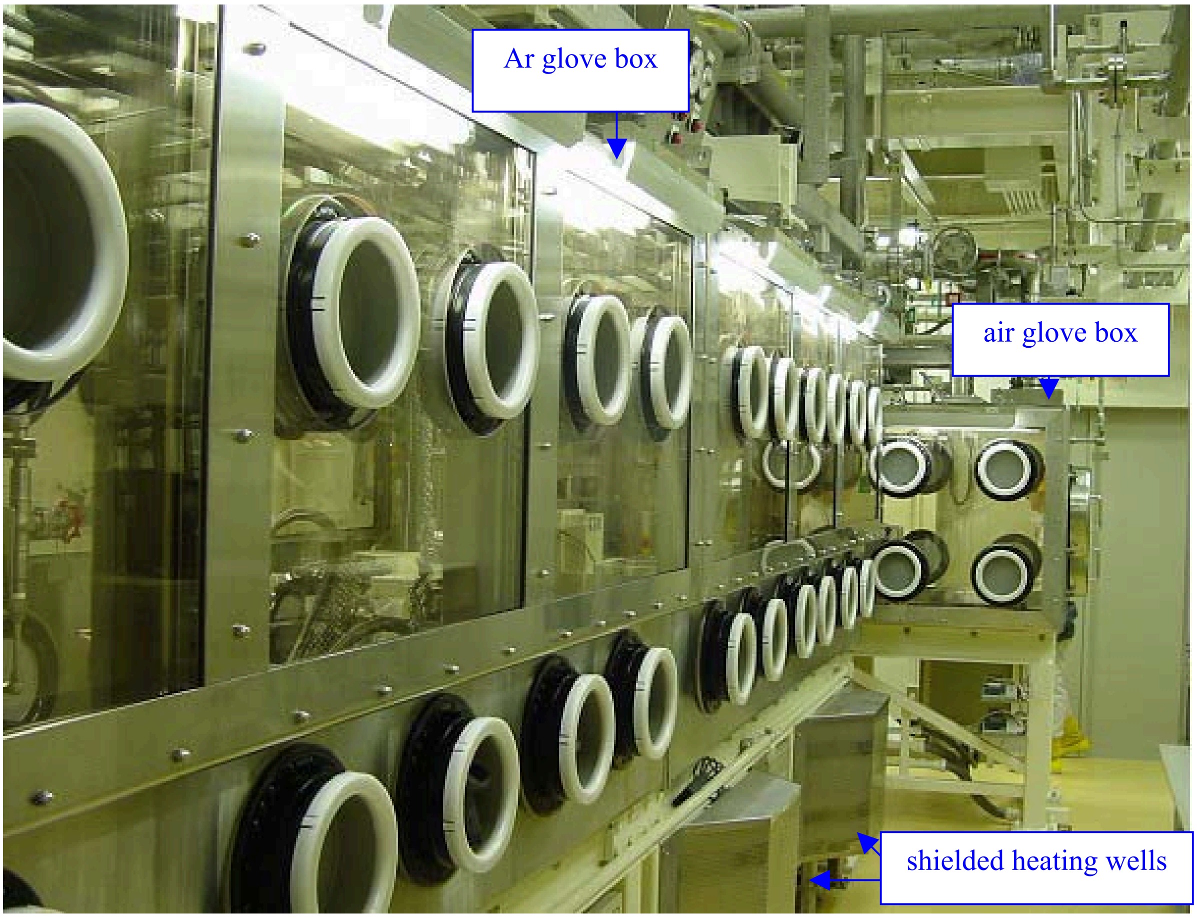


FIG. 8. New glove box system for integrated experiments with electrometallurgical technology.

FIG. 9. Li reduction apparatus.

FIG. 10. Electrorefining apparatus.

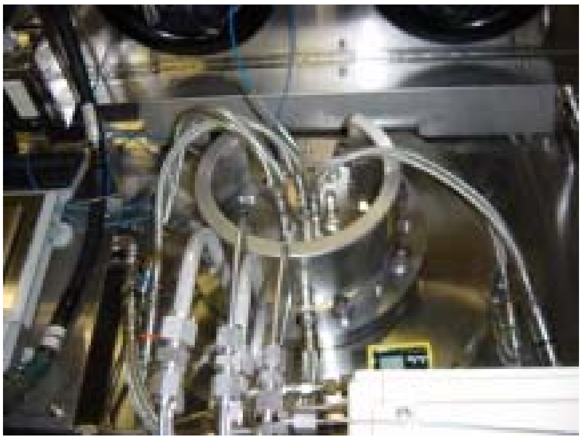


FIG. 11. Cathode processing apparatuss. *FIG. 12. Oxidation apparatus.*

4.3. Integrated test with non-radioactive simulants

As the first step to implement the integrated experiments, non-radioactive materials that simulate actinide behavior in each step were selected. In the course of this experiments, the integrity of experimental procedure as well as the function of apparatus were confirmed as follows.

In the Li reduction process, TiO and Fe_3O_4 were selected for UO_2 simulants. Obtained reduction ratio of TiO in one experiment was about 85%, though Fe_3O_4 was reduced almost completly. The difficulty of reduction of TiO compared with Fe_3O_4 was as expected because of the difference of diffusion coefficient and thermodynamic stability. The complete reduction of actinide oxides is expected in the same experimental condition according to the study of CRIEPI [25,26].

In the electrorefining process, titanium and cerium was selected for U, Pu simulants. Metal cerium was charged into the anode basket, then electro-transported to either solid iron cathode or liquid Cd cathode. Fig. 13 shows the obtained deposit consists of metal cerium and entrained salt. The measured faradaic efficiencies are about 100 % at both electrodes. The reduced TiO obtained in the Li reduction experiment was then charged into the electrorefiner as an anode, and a metallic deposit was recovered at the solid iron cathode.
In cathode process, cerium deposit obtained in the electrorefining process was charged. The entrained salt was distilled off at around 1000^{o}C with leaving cerium metal in the crucible. Distillation of salt-Cd mixture was also examined for simulating Cd cathode deposit. About 100% of salt and Cd were recovered in the condenser-collector as shown in Fig. 14.

In the oxidizing process, metal cerium was oxidized with Ar-O_2 gas at around 500^{o}C. Complete oxidation was confirmed by weight increase and X-ray diffraction.

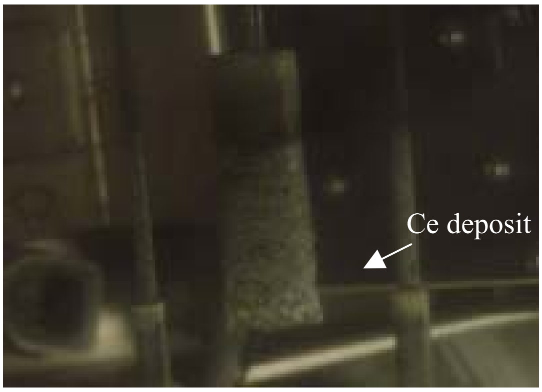


FIG. 13. Solid cathode deposit (Ce).

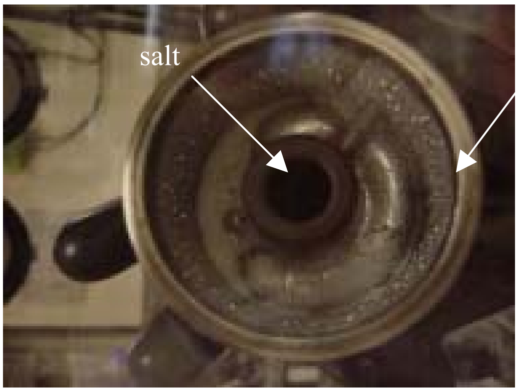


FIG. 14. Recovered salt and Cd by cathode process.

5. CONCLUSION

In order to demonstrate an innovative technology to close actinide cycle, called *electrometallurgical pyro-processing*, CRIEPI and JNC has started new joint study to implement an integrated experiments of electrometallurgical reprocessing of metal and oxide Pu-containing fuel. The new experimental apparatus dedicated for this experiment has been installed in CPF of JNC-Tokai Works. The apparatus is now under internal safety review to start active experiments with actinides such as U and Pu. Integrity of the experimental procedure as well as the function of apparatus was confirmed by the experiments carried out with non-radioactive simulants. After start operation with actinides, this test will enable us to evaluate the technical feasibility of this process to apply closing actinides cycle.

6. REFERENCES

[1] TANAKA, HIROSHI et al., “Design study on advanced reprocessing systems for FR fuel cycle,” GLOBAL2001, September 2001, Paris, 054(2001).

[2] USDOE, TOPS task force of the Nuclear Energy Research Advisory Committee (NERAC), “Technological opportunities to increase the proliferation resistance of global civilian nuclear power systems”, January 2001

[3] CHANG, Yoon I., “The integral fast reactor”, Nuclear Technology, 88, 129(1989).

[4] GARCIA, H.E., et al., “Proliferation resistance of advanced sustainable nuclear fuel cycles”, Proceedings of GLOBAL2001, September 2001, Paris, 299(2001).

[5] INOUE, TADASHI, "Actinide recycling by pyro-process with metal fuel FBR for future nuclear fuel cycle system", Progress in Nuclear Energy, Vol.40, No.3-4, 547-554,2002

[6] GLATZ, JEAN-PAUL, et al., “Advanced reprocessing research at ITU”, Proceedings of GLOBAL2001, September 2001, Paris, 113(2001).

[7] DONALDSON, N., et al., “Pyrochemistry- from flow sheet to industrial facility”, ibid, 193(2001).

[8] BENEDICT, ROBERT et al., “Progress in electrometallurgical treatment of spent nuclear fuel”, J.Nuclear Science and Technology, Supplement 3 (Proceedings of Actinide2001), p749.

[9] KOYAMA, TADAFUMI, JOHNSON, TERRY R., FISCHER, DON F., “Distribution of actinides between molten chloride salt / cadmium metal systems”, J. Alloys and Compounds, 189, 37(1992).

[10] BATTLES, JAMES E., et al., MISHRA, B. ed., “Actinide Processing: Methods and Materials”, The Minerals, Metal & Materials Society, 135(1994).

[11] USAMI, TSUYOSHI et al., “Behavior of actinide elements in the lithium reduction process (2) - Determination of the condition to reduce Am and Np, and evaluation of the reduction behavior of MOX pellet”, CRIEPI Report(in Japanese), T00020, April 2001.
[12] SHIRAI, O. et al., “Recovery of U by electrolysis in LiCl-KCl eutectic melts”, ibid, p745, November 2002.
[13] ACKERMAN, JOHN P., et al., “Treatment of wastes in the IFR fuel cycle”, Progress in Nuclear Energy, 31, 141(1997).
[14] KOYAMA, TADAFUMI, et al., "Waste form development for immobilization of radioactive halide salt generated from pyrometallurgical reprocessing", Proceedings of GLOBAL'97, Yokohama, October 1997.
[15] SIMPSON, M.F., et al., “A description of the ceramic waste form production process from the demonstration phase of the electrometallurgical treatment of EBR-II spent fuel”, Nuclear Technology, 134, 263(2001).
[16] JOHNSON, S.G., et al., “Long-term immersion test results of the metallic waste form from the EMT process of EBR-II spent metallic fuel”, Proceedings of the fifth Topical Meeting on DOE Spent Nuclear Fuel and Fissile Materials Management, Charleston, SC, September 17-20, 2002.
[17] INOUE, TADASHI, et al., “Development of partitioning and transmutation technology for long-lived nuclides”, Nuclear Technology, 69, 206(1991).
[18] KOYAMA, TADAFUMI et al., ”Demonstration of pyrometallurgical processing for metal fuel and HLW”, Proceedings of GLOBAL2001, September 2001, Paris, 170(2001).
[19] TOKIWAI, MORIYASU, et al., “Development of fabrication technology of metallic fuel by injection casting”, J.Nuclear Science and Technology, Supplement 3 (Proceedings of Actinide2001), p910, November 2002
[20] INOUE, Tadashi, et al., “Characterization of fuel alloys with minor actinides”, Trans. Am. Nucl. Soc., 64, 552(1991).
[21] SAKAMURA, YOSHIHARU, et al., “Measurement of standard potentials of actinides (U,Mp,Pu,Am) in LiCl-KCl eutectic salt ans separation of actinides from lanthanides by electrorefining”, J. Alloy. Comp., 271-273, 592(1998).
[22] KOYAMA, TADAFUMI et al.,”Study of molten salt electrorefining of U-Pu-Zr alloy fuel”, J.Nuclear Science and Technology, Supplement 3 (Proceedings of Actinide2001), p765, November 2002.
[23] KOYAMA, TADAFUMI et al., “An experimental study of molten salt electrorefining of uranium using solid iron cathode and liquid cadmium cathode for development of pyrometallurgical reprocessing”, J. Nucl. Sci. Technol., 34, 384(1997).
[24] IIZUKA, MASATOSHI, et al., "Actinides recovery from molten salt/liquid metal system by electrochemical method", *J. Nuclear Materials*, **V247**, 183-190(1997).
[25] USAMI, TSUYOSHI, et al., J. Nucl. Mater., 300, 15(2002).
[26] USAMI, TSUYOSHI, et al., submitted to J. Nucl. Mater.
[27] KURATA, M., et al., “Conversion of high level waste to chloride for pyrometallurgical partitioning of minor actinides”, Proceedings of 7th Int. Conf. On Radioactive Waste Management and Environmental Remediation, ICEM’99, Nagoya, 26-30 September, 1999.
[28] KINOSHITA, KENSUKE, et al., “Separation of uranium and transuranic elements from rare earth elements by means of multistage extraction in LiCl-KCl/Bi system, J. Nucl.Sci.Technol., 36, 189(1999).
[29] NODA, HIROSHI, et al., “Feasibiolity study on commercialized FR cycle system in Japan –The results in the first phase and future plans of the study-“, Proceedings of GLOBAL2001, September 2001, Paris, 052(2001).

[30] TOKIWAI, MORIYASU et al., “Development of metallic uranium recovery technology from uranium oxide by Li reduction and electrorefining”, ibid, p917.
[31] NOMURA, SHIGEO, et al., “Development od challengeable reprocessing and fuel fabrication technologies for advanced fast reactore fuel cycle“, ibid, 042(2001).
[32] KITAWAKI, SHINICHI, et al.,”Evaluation of pyrochemical reprocessing technologies for practical use”, Proc. Workshop on Pyrochemical Separation, 14-16 March, 2000, Avigon, France, p87.

THE DUPIC FUEL CYCLE – RECYCLE WITHOUT REPROCESSING

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Abstract. Since 1991, KAERI, AECL and the United States, with the participation of IAEA, have been engaged in a practical exercise in developing a spent fuel recycle process to extend resources and reduce wastes, while enhancing proliferation resistance over typical recycle options. The concept of the DUPIC fuel cycle is to reuse spent pressurized water reactor fuel as a fuel for CANDU reactors without the reprocessing operations typical of recycling fuel cycles. The basic requirements for the DUPIC fuel cycle development, the fuel fabrication process, the performance evaluation, the development and implementation of safeguards and the factors resulting in enhanced proliferation resistance are described. DUPIC pellets and elements have been sucessfully manufactured at KAERI and AECL for irradiation tests at HANARO and NRU research reactors, respectively. The performance of DUPIC fuel is similar to that of conventional high burnup CANDU fuel, and more extensive work is under way to demonstrate DUPIC fuel perfomance under power reactor conditions.

1. INTRODUCTION

In order to maintain sustainable growth of nuclear energy, it is necessary to develop a fuel cycle technology to improve the uranium resource utilization and to ease the spent fuel management problem by reusing the spent nuclear fuel. The recycling of the spent fuel should be performed with enhanced proliferation resistance throughout the entire fuel cycle. The Generation IV International Forum, which was initiated by the US in 2001 and involves nine additional participating countries, including the Republic of Korea and Canada, has identified “Proliferation Resistance and Physical Protection” as one of four goal areas for the development of future-generation nuclear energy systems that can provide reliable energy products in future. Other goal areas include sustainability, safety and reliability, and economic competitiveness. The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), which was initiated by the IAEA in 2000, selected non-proliferation as a key component for fuel cycles fulfilling energy needs in the 21st century along with economics, environment, fuel cycle waste, and safety.

In order to develop a fuel cycle technology that can utilize spent fuel in an economic way with enhanced proliferation resistance, the concept of the DUPIC (direct use of spent PWR fuel in CANDU reactors) fuel cycle has been proposed to reuse spent PWR (pressurized water reactor) fuel as a fuel for CANDU (Canada deuterium uranium) reactors without the reprocessing operations typical of recycling fuel cycles. Since 1991, KAERI (Korea Atomic Energy Research Institute), AECL (Atomic Energy of Canada Limited) and the United States (Department of State and Los Alamos National Laboratory), with the participation of IAEA, have been engaged in a practical exercise to develop a spent fuel recycle process to extend resources and reduce waste, while enhancing proliferation resistance over typical recycle options.

2. CONCEPT AND BENEFITS OF DUPIC FUEL

The residual content of fissile materials in spent PWR fuel is about 1.5 wt% of U-235 and Pu-239 combined, for a discharge burnup of 35,000 MWd/MTU. Since the CANDU reactor was designed to use natural uranium oxide fuel with 0.7 % U-235, it has excellent neutron economy that allows it to use spent PWR fuel directly without a reprocessing stage, even though it contains fission products and transuranic elements. DUPIC fuel burnup is expected to be about 15,000 MWd/MTHE, which is twice that of natural uranium fuel at 7,000 MWd/MTU. For the case of PWR discharge burnup of 35,000 MWd/MTU, the optimum ratio of PWR to CANDU reactors is 2.5, which means the spent PWR fuel from 2.5 PWRs can supply DUPIC fuel for one CANDU reactor.

The main characteristics of the DUPIC fuel cycle technology are the enhancement of the proliferation resistance for reusing spent nuclear fuel without removing all the fission products, simplicity of the fuel treatment process and a great reduction of radioactive process waste owing to inherent features of the direct dry process. The implementation of the DUPIC fuel cycle will provide multiple benefits, such as:

(i) eliminating accumulated PWR spent fuels by burning them in CANDU reactors ,
(ii) reducing CANDU spent fuel arising owing to the higher burnup of the DUPIC fuel, and
(iii) savings of natural uranium resources required to produce CANDU fuel.

3. DUPIC FUEL MANUFACTURING TECHNOLOGY DEVELOPMENT

The DUPIC fuel fabrication process was developed using the OREOX (oxidation and reduction of oxide fuel) process to prepare resinterable fuel stock material from spent PWR fuel and adopting the powder/pellet route to fabricate DUPIC fuel. The spent PWR fuel assembly is first disassembled and decladed mechanically to take the spent PWR fuel materials out of the spent fuel assembly. The key process is to reduce the spent fuel materials into a resinterable powder by the OREOX process. Once the resinterable powder feedstock is prepared, the remaining fabrication steps are similar to conventional processes, such as precompaction, granulation, final compaction and sintering. The element seal welding is performed by Nd:YAG laser, which is supplied from outside the hot cell by optical fiber. The fuel fabrication flow sheet is shown in Figure 1. All the fuel fabrication processes are remotely conducted in concrete hot cells due to the high radioactivity that results because there are no process steps that remove the fission products and transuranic elements from the spent fuel material. Remote fuel fabrication equipment was developed and installed in IMEF (Irradiated Material Examination Facility) hot cell facilities at KAERI in 2000.

The optimum conditions of oxidation and reduction were determined to be 450 ^{o}C in air for 1 hour and 700 ^{o}C in 4 % H_2-Ar for 4 hours, respectively. Three cycles of oxidation and reduction steps with subsequent milling of the OREOX treated powder have shown that it can produce powder feedstock to manufacture sound sintered pellets with a density of greater than 95% of the theoretical density (TD). The powder morphology observed by SEM is shown in Figure 2 for the various OREOX treatments. KAERI performed a series of remote fabrication campaigns to produce DUPIC fuel pellets and elements. The fabricated DUPIC fuel using spent PWR fuel discharged from Gori Nuclear Power Plant in 1986 satisfies current fuel specifications such as fuel density (%TD), grain size and pellet integrity, etc. The OREOX treated powder and the sintered pellets remotely fabricated are shown in Figure 3. These results from KAERI experiments were able to successfully duplicate a similar process used by AECL to successfully fabricate three DUPIC elements at the Whiteshell Laboratories in 1997 using spent BWR fuel.

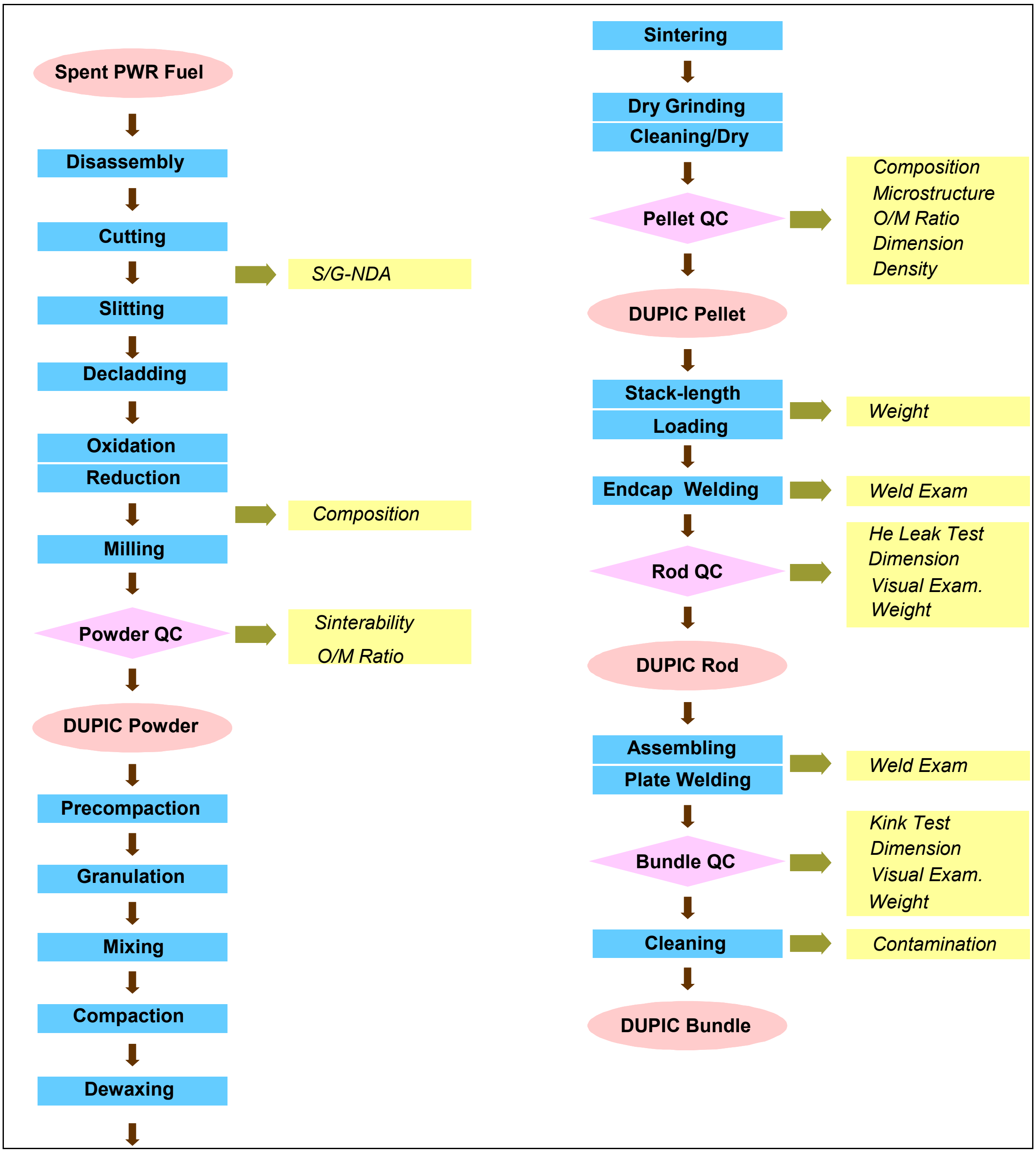

FIG. 1. DUPIC fabrication flow.

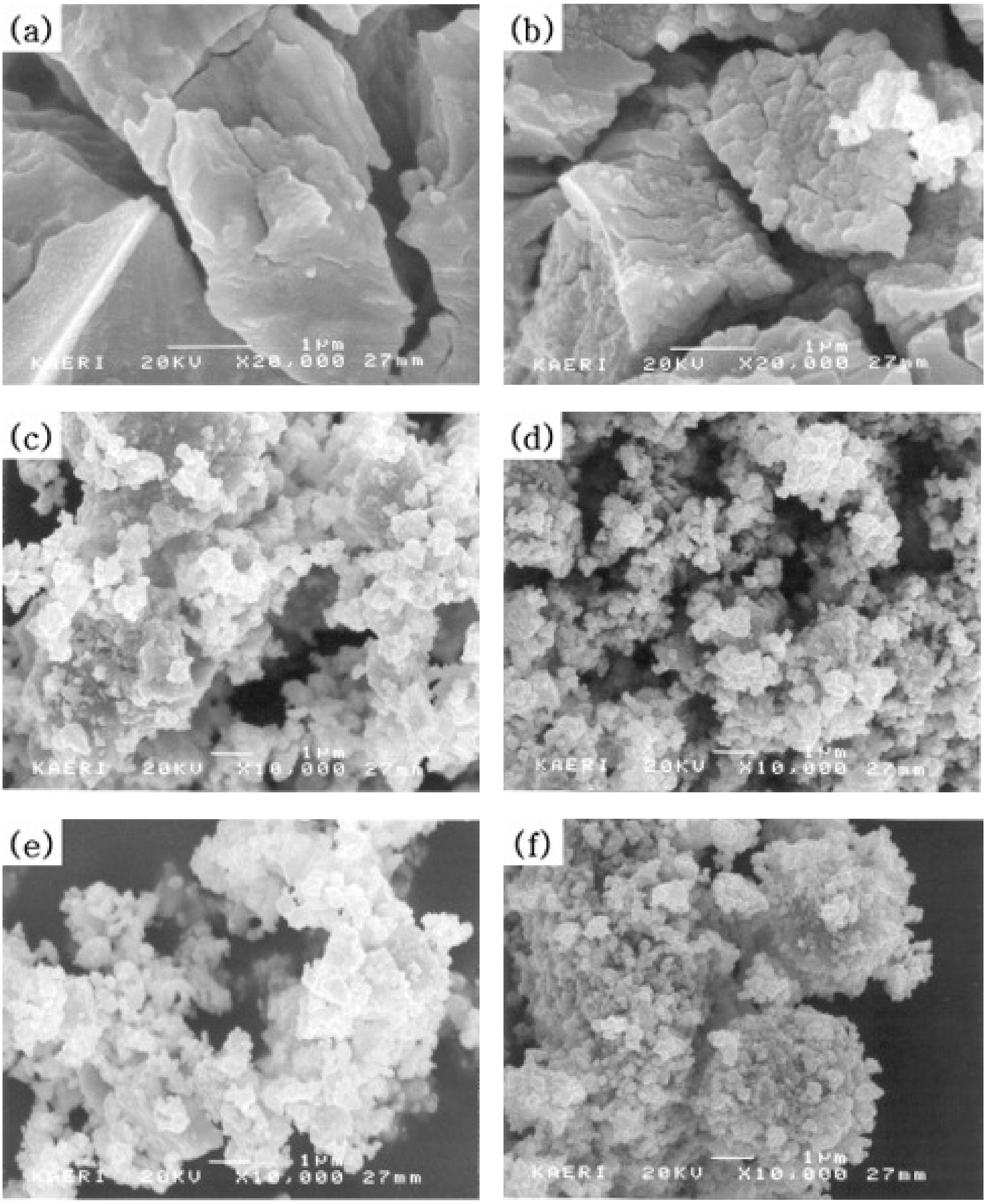

FIG. 2. Morphology of OREOX and Milling Treated Powders : (a) 1 cycle OREOX, (b) 3 cycles OREOX, (c) 1 cycle OREOX and 15 min milling, (d) 3 cycles OREOX and 15 min milling, (e) 1 cycle OREOX and 120 min milling, and (e) 3 cycle OREOX and 120 min milling.

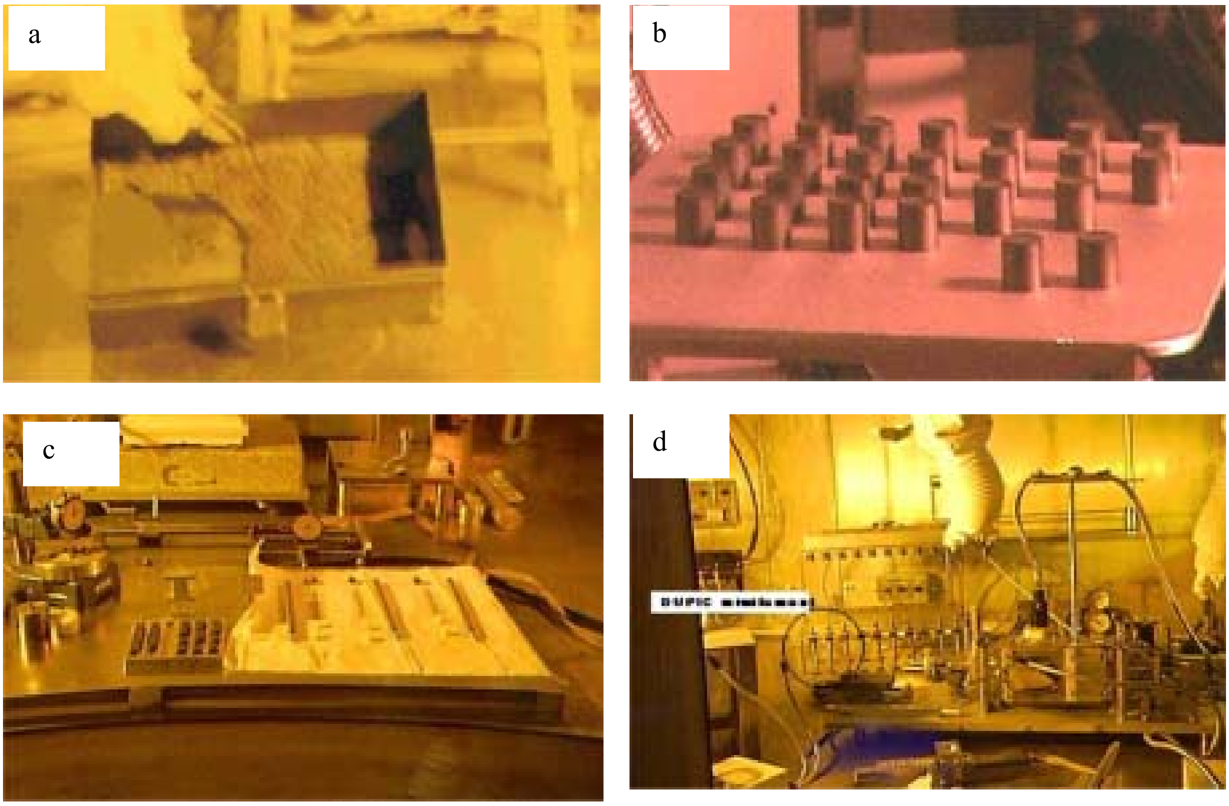

FIG. 3. Remotely fabricated DUPIC fuel : (a) DUPIC powders after OREOX process, (b) sintered pellets, (c) inspection of DUPIC pellets, and (d) DUPIC element welding.

4. PERFORMANCE EVALUATION OF DUPIC FUEL

DUPIC fuel pellets are expected to have lower thermal conductivity and higher centerline temperature than fresh uranium dioxide because of their higher content of fission products. In order to verify the performance of DUPIC fuel experimentally, a series of irradiation tests are currently under way using the HANARO research reactor in KAERI and the NRU research reactor in AECL. The typical microstructure of the irradiated DUPIC pellets irradiated at HANARO up to a burnup of 6,700 MWd/MTHE with linear element rating of 36 KW/m is shown in Figure 4. A detailed post-irradiation examination is under way on the KAERI fuel. DUPIC elements fabricated by AECL were also irradiated in AECL's NRU fuel test loops to 10,000 and 16,000 MWd/MTHE under conditions simulating those in a CANDU fuel channel. From post-irradiation examination it was found that the performance of DUPIC fuel with a power history peaking at 45 kW/m early in life, and declining thereafter was found to be similar to fresh uranium dioxide irradiated under similar conditions, except that the DUPIC fuel experienced more pellet-centre microstructural changes and slightly higher fission gas release. This result is acceptable and consistent with expectations, given the high total burnup of the fuel material (28,000 MWd/MTHE in a light water reactor, plus an additional 16 MWd/MTHE as DUPIC fuel). In addition, performance analysis using the performance code such as ELESTRES has been attempted. Material properties for the analysis such as thermal conductivity, thermal expansion, creep and Young's modulus, etc. were measured using simulated fuel. The results of irradiation tests and predictions of the performance code will be analyzed for verification of the DUPIC fuel performance.

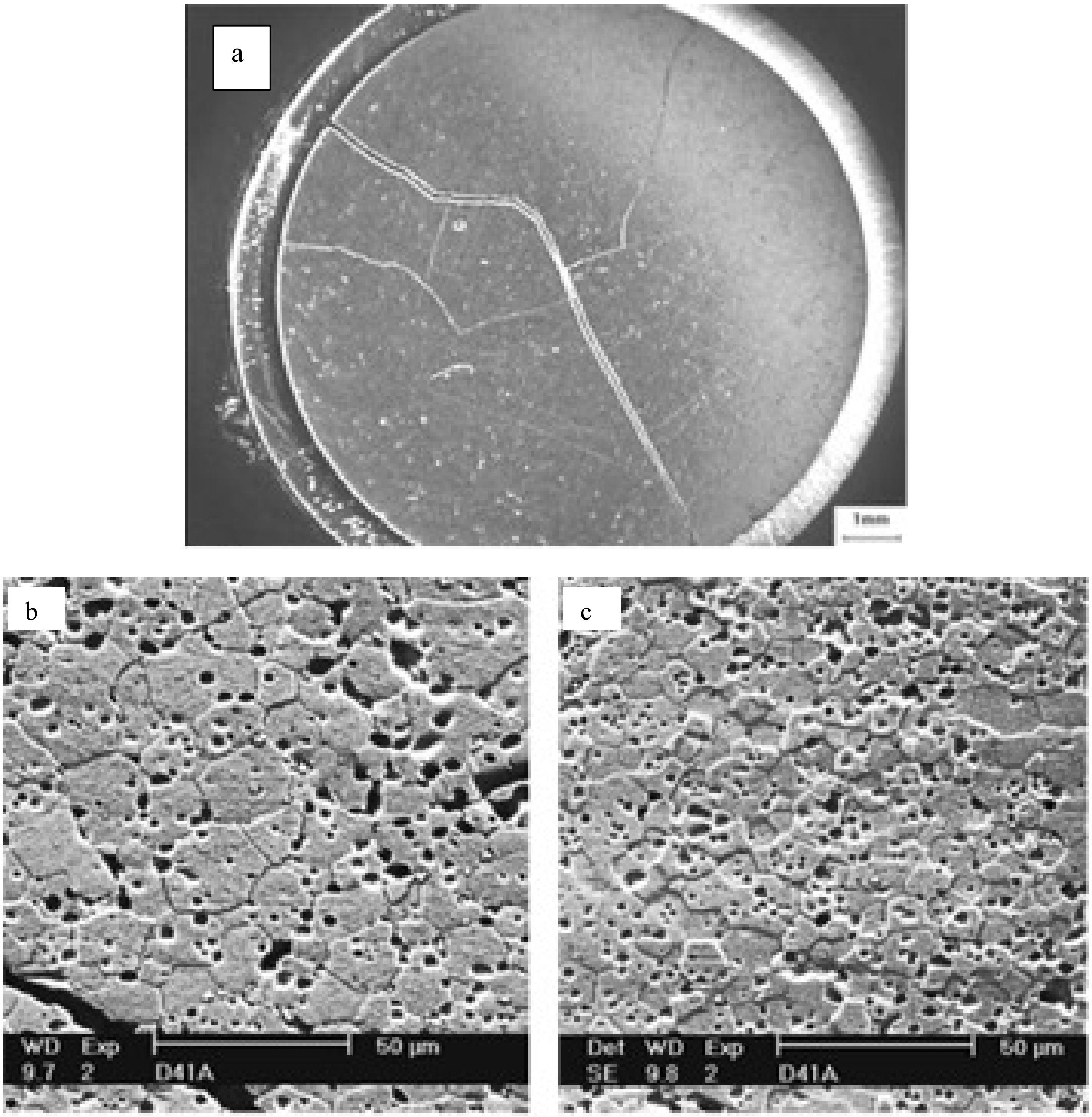

FIG. 4. Post-irradiation examination of irradiated DUPIC fuel: (a) cross-sectional optical micrograph of DUPIC fuel, (b) SEM micrograph of central region of DUPIC fuel pellet, and (c) SEM micrograph of rim region of DUPIC fuel pellet.

In order to verify the compatibility of DUPIC fuel with current CANDU reactors, modification of the fuel design and refueling operation are recommended. The on-power refueling scheme of DUPIC fuel would be modified from the normal 4-8 bundle shift used for natural uranium to a 2 bundle shift due to the fuel's higher fissile content. Although the CANDU reactor is capable of safely utilizing DUPIC fuel without modifications, it is also possible to redesign the fuel bundle to contain a neutron poison (such as dysprosium) to reduce the reactivity during postulated accident scenarios involving loss of primary coolant. Since DUPIC fuel is fabricated from spent PWR fuel with inhomogeneous compositions due to variations in discharge burnup, care must be used to judiciously mix fuel from PWR assemblies such that the final blend meets the reactivity criteria. It is possible that an adjustment of fissile contents of DUPIC fuel may be required using a small quantity of fresh fuel. It has been confirmed that most of the requirements for nuclear, thermal-hydraulic and mechanical compatibility can be satisfied using DUPIC fuel.

5. SAFEGUARDS TECHNOLOGY DEVELOPMENT

As is typical for bulk handling facilities, safeguards technology development for DUPIC fuel fabrication is focused on the management of accurate accountability in process inventory to maintain knowledge of continuity in material flow, and ensure that the DUPIC process within the containment and surveillance system is able to give assurance of non-diversion of material.

The overall DUPIC process materials balance is monitored by Cm-244 measurement using a neutron coincidence counting method. The principle is to use the spontaneous fission neutron emissions from Cm-244 to quantify the plutonium contents of fuel materials. Because there is no chemical separation involved in the DUPIC fuel cycle, the ratio of Cm-244 to plutonium should be constant at the input, output, and process steps of the DUPIC fuel cycle. Therefore, it is possible to establish the associated plutonium inventory by knowing the Cm-244-to-plutonium ratio and a measured value of Cm-244. Based on this concept, a coincidence neutron counter, called DSNC (DUPIC Safeguards Neutron Counter), has been developed by KAERI and LANL as shown in Figure 5. The performance test was successfully completed by the IAEA for use of nuclear material accounting in 1999. In addition to the material accounting system, an unattended continuous monitoring system with diagnosis software for the data acquisition, review and data evaluation based on a neural network system is being developed by KAERI jointly with LANL.

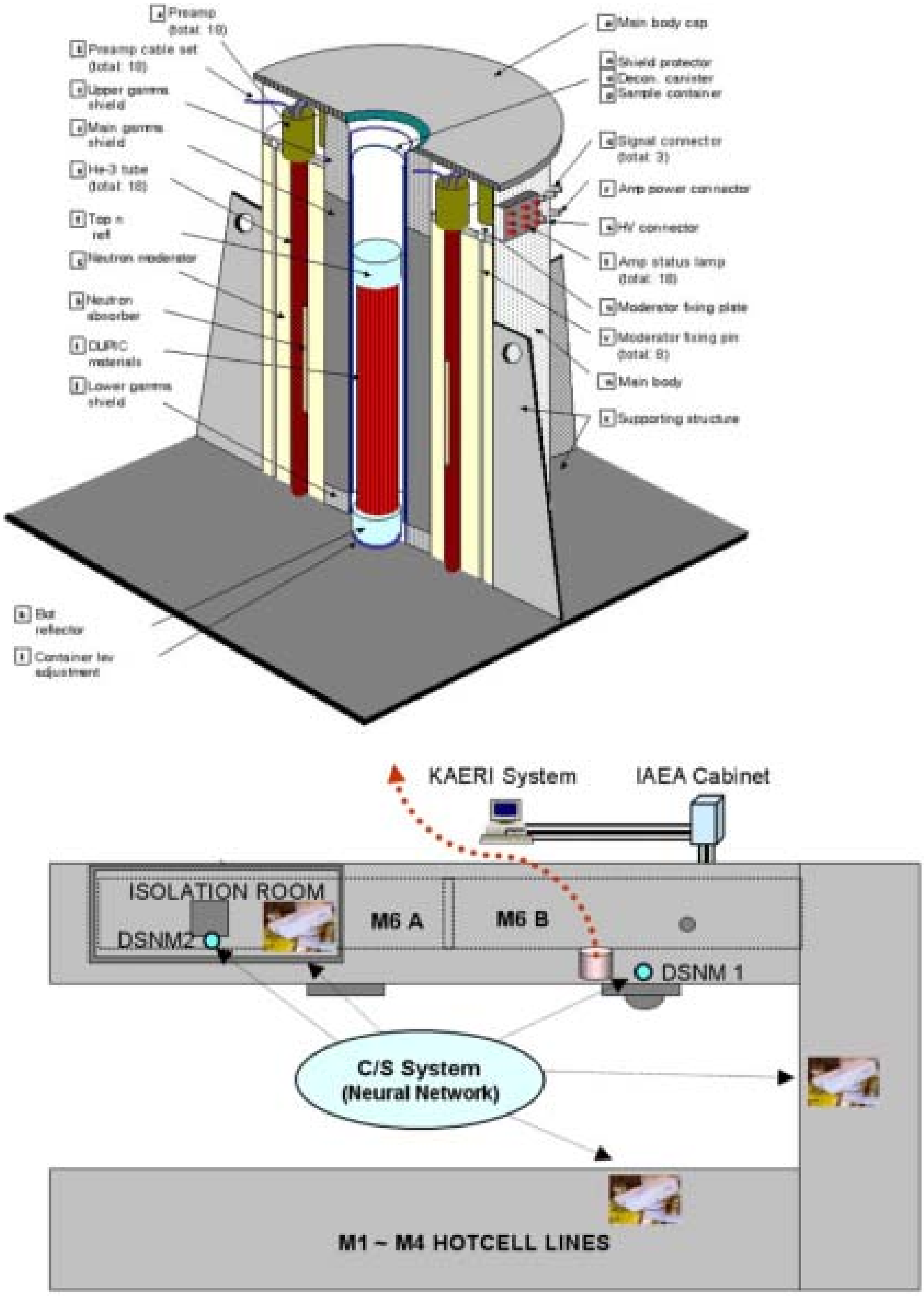

FIG. 5. DSNC (Curium Boy) for nuclear material accounting.

6. PROLIFERATION RESISTANCE

The DUPIC fuel cycle contains a number of intrinsic features enhancing proliferation resistance. There is no production of an unirradiated direct use material. The fuel remains highly radioactive throughout the process. This results in part from the intentional decision to avoid undertaking any step intended solely for intentional fission product separation. There is coincidental separation of volatiles and semi-volatiles that occurs during the oxidation and reduction steps, but this does not fundamentally alter the suitability of the material for nuclear explosive use. The other major factor in maintaining the radioactive nature of the process is the OREOX process itself is incapable of producing separated plutonium and the process cannot be readily modified to do so. This characteristic was recognized in the early studies of oxidation-reduction processing. Early characterizations of these processes as reprocessing clearly misstate their nature and are inappropriate. The IAEA is applying the safeguards criteria for spent fuel handling to the DUPIC fuel cycle activities, not those for a reprocessing plant, because of the absence of any unirradiated direct use material. In sum, the characteristics of the materials produced during the DUPIC fuel cycle are unusable for nuclear explosives without substantial additional processing of the type normally associated with reprocessing. All fuel cycle steps must be conducted remotely because of the radioactive nature of the material. The requirement for substantial shielding is an intrinsic feature that contributes to the safeguardability of the fuel cycle.

The fact that DUPIC fuel development is being conducted in Korea and Canada is also significant from the point of view of proliferation resistance. The requirement to pursue a fuel cycle with enhanced intrinsic proliferation resistance flows from Korean national fuel cycle policies adopted due to the political situation on the Korean peninsula. Nevertheless, there are also substantial intrinsic proliferation resistance features present in this fuel development. Korea and Canada are both parties to the Nuclear Non-Proliferation Treaty. Both have full scope safeguards agreements in effect. The fuel involved is subject to nuclear cooperation agreements with the United States that establish conditions on its use. In Korea, IAEA safeguards are supplemented by a national inspection system. The importance of such intrinsic features was highlighted in a recent IAEA report on proliferation resistance.

7. INTERNATIONAL COOPERATION

In order to maintain the transparency of the research on handling and processing spent nuclear fuel and to efficiently make use of available expertise, an international cooperation framework has been in place from the beginning of the DUPIC technology development. The feasibility study was performed during 1991 to 1993 among KAERI, AECL and LANL of the United States. Since 1994, the development of DUPIC fuel and the experimental verification of its performance have been conducted by KAERI and AECL to demonstrate the technical and economic viability of the fuel cycle. Safeguards technology has been developed by KAERI and LANL. The IAEA has participated in the program for the purpose of identifying safeguards questions and to ensure that the process that is developed is safeguardable. To the knowledge of the authors, this is the earliest involvement of the IAEA in any fuel cycle development program.

8. REFERENCES

[1] Fundamentals of Proliferation Resistance, Como, Italy, October 28-31, 2002.
[2] Technical Aspects of Enhancing Proliferation Resistance, Vienna, Austria, November 19-21, 2002.

[3] YANG, M.S., LEE, Y.W., BAE, K.K., NA, S.H., "Conceptual study on the DUPIC fuel manufacturing technology", Proc. Int. Conf. and Technology Exhibition on Future Nuclear System, GLOVAL '93, Seattle, 1993
[4] LEE, J.W., KIM, W.K., KIM, S.S., YANG, M.S., PARK, H.S., "DUPIC fuel fabrication in shielded facilities in Korea", Proceedings of the 7th CANDU Fuel Conference, Kingston, Canada, 2001, pp227
[5] KO, W.I., KIM, H.D., YANG, M.S., "Advantages of irradiated DUPIC fuels from the perspective of environmental impact", Nuclear Technology Vol.138, May 2002, pp123.
[6] GRANTHAM, L.F., CLARK, R.G., HOYT, R.C., and MILLER, J.R., "AIROX dry pyrochemical processing of oxide fuels: a Proliferation-resistant Reprocessing Method", Rockwell International report, 1980; also in Am. Chem. Soc. Symposium Series 117, 219, 1980
[7] International Atomic Energy Agency, Proliferation Resistance Fundamentals for Future Nuclear Energy Systems, STR-332, December 2002

ASSESSING THE PROLIFERATION RESISTANCE OF INNOVATIVE NUCLEAR FUEL CYCLES

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Abstract. The National Nuclear Security Administration is developing methods for nonproliferation assessments to support the development and implementation of U.S. nonproliferation policy. This paper summarizes the key results of that effort. Proliferation resistance is the degree of difficulty that a nuclear material, facility, process, or activity poses to the acquisition of one or more nuclear weapons. A top-level measure of proliferation resistance for a fuel cycle system is developed here from a hierarchy of metrics. At the lowest level, intrinsic and extrinsic barriers to proliferation are defined. These barriers are recommended as a means to characterize the proliferation characteristics of a fuel cycle. Because of the complexity of nonproliferation assessments, the problem is decomposed into: metrics to be computed, barriers to proliferation, and a finite set of threats. The spectrum of potential threats of nuclear proliferation is complex and ranges from small terrorist cells to industrialized countries with advanced nuclear fuel cycles. Two general categories of methods have historically been used for nonproliferation assessments: attribute analysis and scenario analysis. In the former, attributes of the systems being evaluated (often fuel cycle systems) are identified that affect their proliferation potential. For a particular system under consideration, the attributes are weighted subjectively. In scenario analysis, hypothesized scenarios of pathways to proliferation are examined. The analyst models the process undertaken by the proliferant to overcome barriers to proliferation and estimates the likelihood of success in achieving a proliferation objective. An attribute analysis approach should be used at the conceptual design level in the selection of fuel cycles that will receive significant investment for development. In the development of a detailed facility design, a scenario approach should be undertaken to reduce the potential for design vulnerabilities. While, there are distinctive elements in each approach, an analysis could be performed that utilizes aspects of each approach.

1. INTRODUCTION

The National Nuclear Security Administration (NNSA) is developing methods for nonproliferation analyses to support the development and implementation of U.S. nonproliferation policy. A Nonproliferation Assessment Methodology (NPAM) Working Group, comprised of representatives from the DOE laboratories and academia, was established to prepare guidelines [1] for the selection of methods and for the performance of nonproliferation assessments.

NNSA directed the Working Group to place initial emphasis on fuel cycle related issues because of a recognized need to support DOE initiatives related to new reactor designs. However, the methods have broad applicability to a spectrum of proliferation issues.

2. OBJECTIVES OF NONPROLIFERATION ASSESSMENT

NNSA provides technical input to policy makers that establish U.S. nonproliferation policies. Within the context of policy development, the purpose of a nonproliferation assessment is to examine the relative merits of alternative actions or propositions relative to controlling the potential for proliferation. For example, as illustrated in Figure 1, the purpose of the analysis may be to compare the proliferation resistance of Fuel Cycle Option A with Fuel Cycle Option B. In order to satisfy the objectives of the analysis, the analyst needs measures of proliferation resistance to indicate whether A is better or worse than B. The analyst also needs to know the uncertainty in the measures to be able to determine whether the indicated difference between A and B is significant. Thus, as indicated in Figure 2, the comparison between Fuel Cycle Option A and Fuel Cycle Option B may have to be interpreted within the context of the uncertainty. Whereas the policy maker may conclude from examining Figure 1 that Fuel Cycle A is more proliferation resistant than Fuel Cycle B, the more appropriate conclusion may be the one drawn from Figure 2 that Fuel Cycle A is probably more proliferation resistant than Fuel Cycle B.

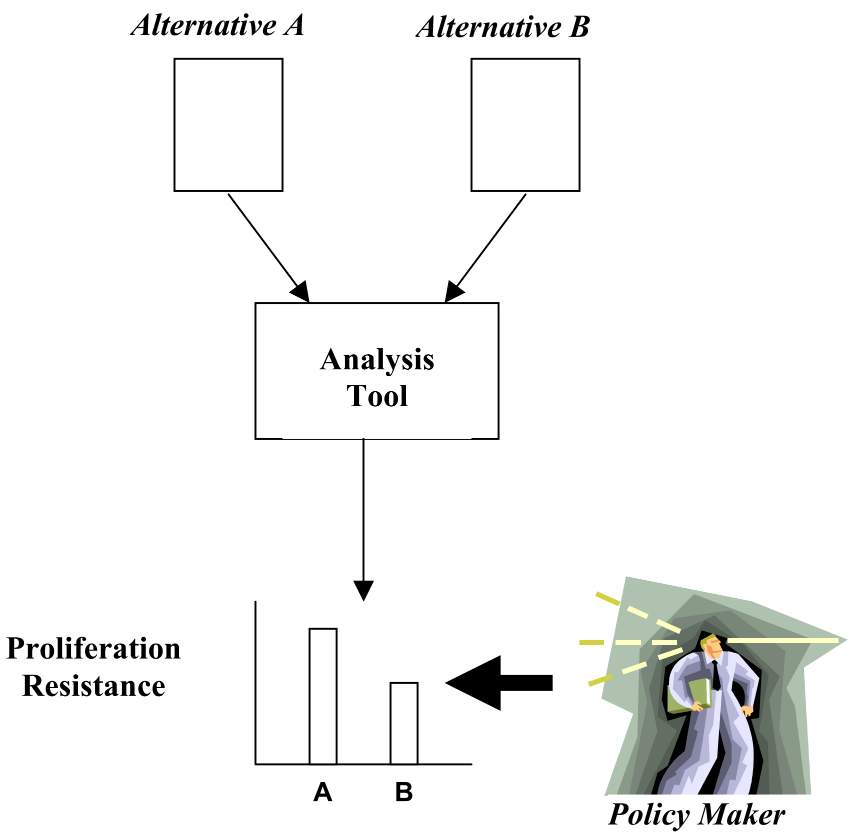

FIG. 1. Concept of a nonproliferation assessment study.

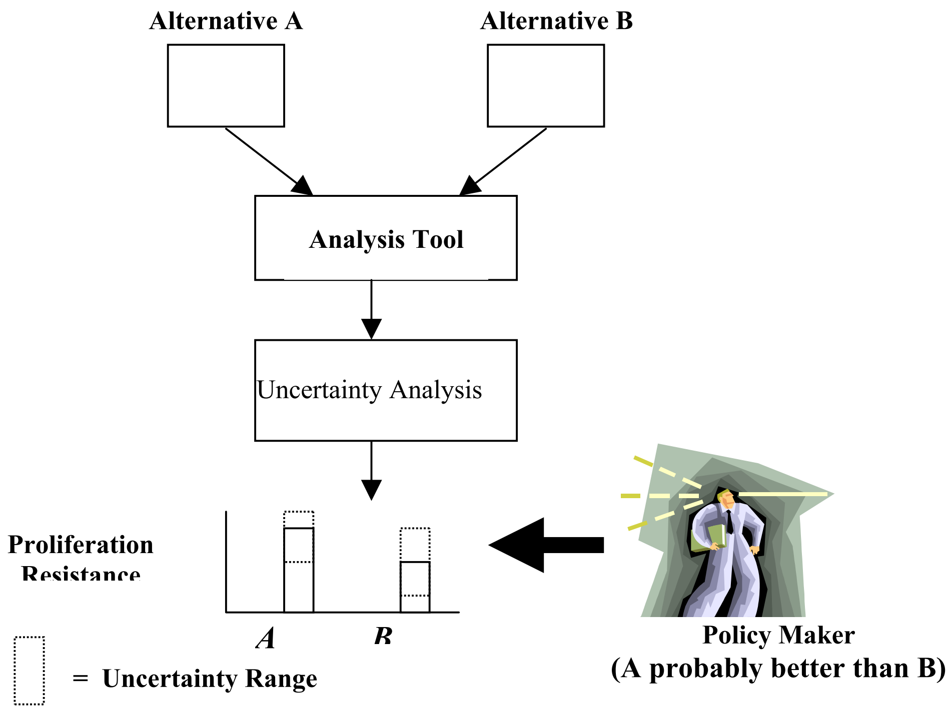

FIG. 2. Results with uncertainties.

3. CONTENT OF GUIDELINES DOCUMENT

Figure 3 illustrates the steps to be taken in performing a nonproliferation assessment in support of policy development. Each step is addressed in the guidelines document. Within the first step of framing the problem, the analyst in collaboration with the policy-maker must clearly state the objectives of the analysis. The analyst must then develop an approach to attack the problem. Based on their experience, nonproliferation analysts have developed some preferred approaches to decomposing nonproliferation problems involving selection of analysis methods, threat description, identification of barriers to proliferation, the definition of metrics, and system segmentation.

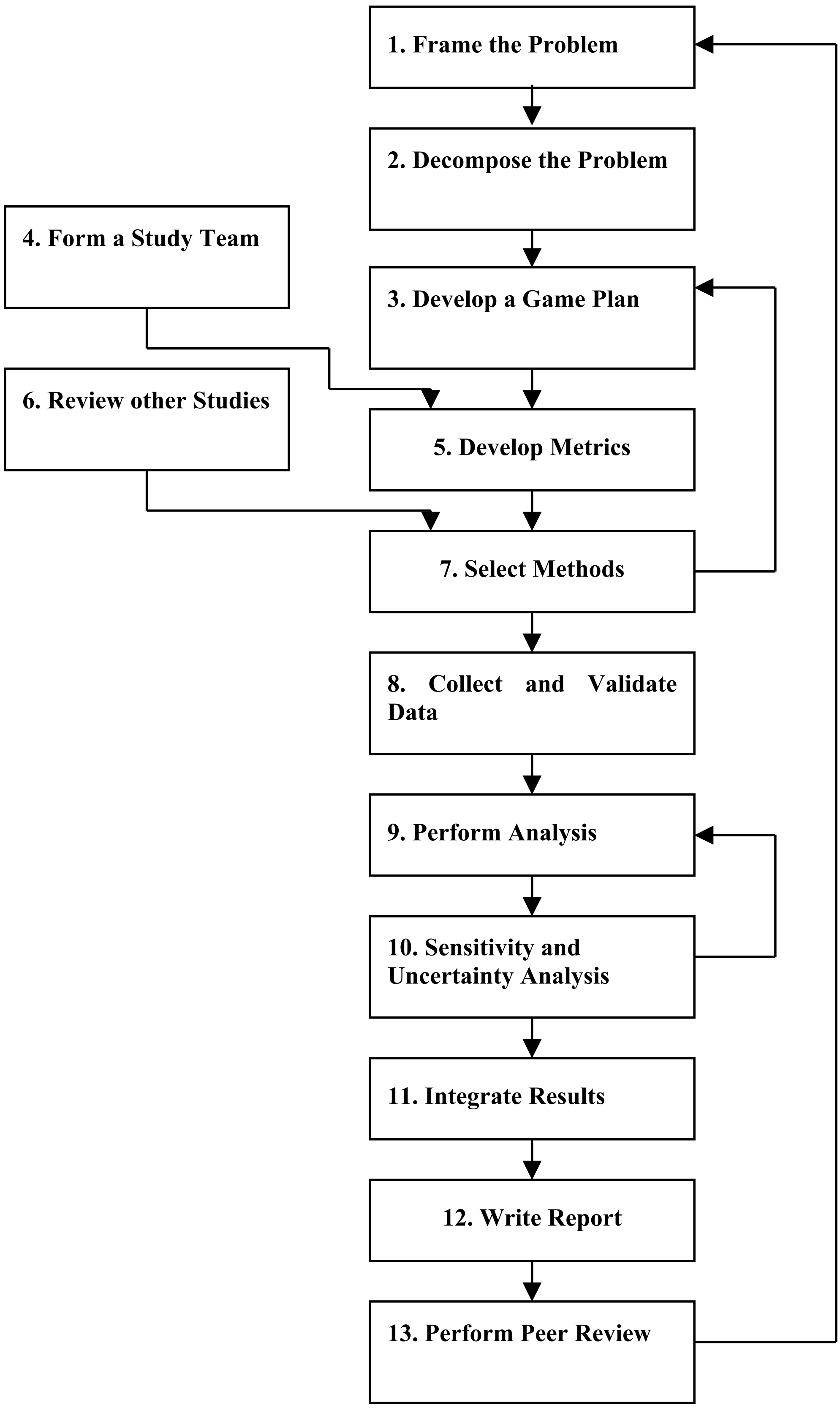

FIG. 3. Elements of an assessment project.

4. METHODS OF ANALYSIS

One of the objectives of the guidelines document is to provide a toolbox of tools and methods that are available to the analyst to support the performance of a nonproliferation assessment. The applicability of a variety of analytical methods is discussed in the guidelines document, including some generic tools that could be applied in any study such as uncertainty analysis, sensitivity analysis, logic diagrams, and approaches to expert elicitation.

Two general categories of methods have been used historically as the basis for nonproliferation assessments: attribute analysis and scenario analysis.

Attribute analysis. In this approach, attributes of the systems being evaluated (often fuel cycle systems) that affect their proliferation potential are identified. For a particular system under consideration, the attributes are weighted subjectively. Typically, these studies are more qualitative than the scenario analysis studies. There is an extensive history of the use of formal methods of decision theory (such as multi-attribute utility theory) to assist in policy development using this type of approach.

Scenario analysis. In these studies, hypothesized scenarios of pathways to proliferation are examined. The analyst models the process undertaken by the proliferant to overcome barriers to proliferation and estimates the likelihood of success in achieving a proliferation objective. Typically, these studies use logic modeling techniques (often probabilistic techniques). The results are quantitative but rely, in some respects, on subjective judgments of experts.

The Working Group identified two-sided methods as also having significant potential to support nonproliferation assessment.

Two-sided methods. These methods examine the interplay between opponents. Wargaming is a two-sided approach that has been used extensively in other applications. A wargame [2] is a role-playing exercise where human participants, often with opposing goals, make sequential decisions to allow a scenario to unfold. Wargames appear to have promising potential to provide policy insights for nonproliferation issues that are not addressed effectively by other methods.

These different categories of methods have complementary roles in the development of nonproliferation policy. Either attribute analysis or scenario analysis can provide the basis for an integrated nonproliferation assessment that assesses measures of the proliferation potential of alternative systems. These two types of approaches have strengths and weaknesses that determine the type of study for which each is preferred. Wargames are not competitive with attribute analysis or scenario analysis for the performance of integrated assessments. The results of a wargame represent a possible outcome but not necessarily the most likely outcome of an interaction. They can, however, provide valuable insights into the thought processes of adversaries as input to an integrated assessment.

The types of policy issues for which NNSA provides technical support can be addressed by the following combinations of nonproliferation assessment characteristics:

— System – materials, facilities, processes, safeguards
    - Commercial fuel cycle
    - Nuclear weapons infrastructure
— Design level
    - Detailed

    - Conceptual
- Location
    - Domestic
    - International – generic
    - International – regional specific

In addition, geopolitical analyses may be required to support “regional security” related policies.

In evaluating the strengths and weaknesses of the different analysis approaches, the Working Group concluded that the principal discriminating features of the assessment are the detail of design information and the amount of time (and associated effort) available to perform the study. Table I summarizes the strengths and weaknesses of the analysis approaches for these different analysis constraints.

High (H), Medium (M) and Low (L) represent the strength of the analysis approach for the indicated level of design detail and project duration. The dividing point between a short duration and a long duration project is assumed to be one month. Sometimes requests for policy input can be as short as one day. Unless there is substantial information already available on the topic, it is difficult to use any of these methods in a period of less than one week. An in-depth, integrated assessment is expected to require approximately six months.

Scenario analysis methods can be applied at a conceptual level but their strength is in studies in which detailed design information is available to support the analysis. This type of assessment requires substantial time and effort. Scenario analysis methods can be important design tools that are used in the identification and correction of design vulnerabilities.

Attribute analysis methods do not require the level of detailed design information as scenario analysis methods. They can, however, require substantial effort. The Analytic Hierarchy Process [3], an attribute method, can be used for quick response projects. However, the reliability of the result depends highly on the expertise of the experts that provide input to the process.

Some time and effort is required to set up a wargaming exercise. However, the actual execution of the exercise is short. Thus, it is quite practical to obtain meaningful results in less than one month. Wargames tend to provide insights that are of value in developing a better understanding of the conflict between the proliferant and the act. Wargames are usually only one element of a larger nonproliferation assessment.

Table I. Strengths and Weaknesses of Assessment Approaches as a Function of Analysis Constraints

System	Location	Design Detail / Duration	Attribute Analysis	Scenario Analysis	War-gaming*
Nuclear Fuel Cycle or Nuclear Weapons Infrastructure	Domestic, Foreign-Generic, or Region Specific	Detailed / Short	M	M	L
		Detailed / Long	H	H	M
		Conceptual / Short	H	L	M
		Conceptual / Long	H	H	H
Geopolitical	Global or Regional	Short	M	L	H
		Long	M	L	H

*Wargaming typically is only one element of a nonproliferation analysis.

5. BARRIERS TO PROLIFERATION

One of the strategies that is typically taken in nonproliferation analysis is to identify barriers to proliferation and to determine how effective these barriers are to deterring proliferation. This strategy is used both by scenario-based and attribute-based approaches. However, the manner in which they assess the effectiveness of barriers differs.

Barriers are typically characterized as either intrinsic, features that are inherent to a particular fuel cycle system, or extrinsic, administratively-added security features such as physical protection and international safeguards. A listing of barriers is provided in Table II. Also included in the table are the attributes of that barrier that affect proliferation potential. The nature of the proliferation threat can impact the relative effectiveness of intrinsic and extrinsic barriers. If a nation state decides to remove its facilities from IAEA safeguards and to use its commercial nuclear facilities to produce weapons material, extrinsic barriers would become completely ineffective in deterring the production of weapons material but intrinsic barriers could still be in place.

6. THREAT DESCRIPTION

Another standard strategy for the decomposition of nonproliferation problems is to define a set of threats and to evaluate the proliferation resistance of the option under consideration for each threat separately. Consider, for example, a fuel cycle facility that is under IAEA safeguards. One threat could be a country with a high level of technical competence that decides to divert material covertly. Another threat is a small subnational group that attacks the facility and attempts to escape with weapons material. The relative resistance to these different proliferation threats varies depending on the alternative fuel cycle system under consideration. Table III provides a possible characterization of threat categories.

7. METRICS

After the objectives of the study have been clearly defined, the analyst must determine the metrics or measures (high level metrics) that will be used to characterize the proliferation resistance of the alternatives being evaluated. The guidelines review metrics that have been used in previous studies. A general hierarchy of metrics, as illustrated in Figure 4, is developed to show how lower level metrics can be related to the high level measures that will be used by the decision maker to decide which are the preferred alternatives. For nonproliferation studies that compare the proliferation characteristics of one fuel cycle with an alternative fuel cycle, the analyst should develop high-level measures that are representative of the characteristics of the fuel cycle or part of a fuel cycle, rather than mixing characteristics of the fuel cycle and the proliferator. The analyst should also develop metrics for evaluation in a manner to minimize dependencies between the metrics as they affect the high level measures. A typical top-level measure is either proliferation resistance, which is a characteristic of a fuel cycle system, or proliferation risk, which also includes aspects of the proliferator.

Table II. Proliferation barriers

Barrier type	Barrier	Attributes
Material barriers	Isotopic	Critical mass Degree of isotopic enrichment Spontaneous neutron generation Heat generation rate Difficulty presented by radiation to weapons design
	Chemical	Degree of difficulty in refining weapons material
	Radiological (dose to humans)	Degree of remote handling normally required
	Mass and bulk	Concentration of material, ease of concealment
	Detectability	Degree of passive detection capability Active detection capability Hardness of radiation signature Uniqueness of material's signatures Uncertainties in detection equipment
Technical barriers	Facility unattractiveness (degree of difficulty of production of weapons material inherent in a facility)	Complexity of required modifications Cost of modifications Safety implications of modifications Time required to modify Facility throughput Effectiveness of observable environmental signatures
	Facility accessibility	Difficulty and time to perform operations Need for specialized equipment Manual versus automatic, remote operation Frequency of operational opportunity to divert
	Available mass	Amount of potentially weapons useable material at a given point in a fuel cycle
	Diversion detectability	Type of material and processes with respect to accountability Uncertainties in detection equipment Form of material as amenable to counting
	Skills, expertise and knowledge	Dual-use skills and knowledge Applicability of dual-use skills Availability of dual-use information
	Time	Time materials in a facility or process are available to proliferator access
Extrinsic (Institu-tional) barriers	Safeguards	Availability and access to information Minimum detectabilty limits for material Ability to detect illicit activities Response time of detectors and monitors Precision and frequency of monitoring Degree of incorporation into process design and operation
	Access control and security	Administrative steps for access Physical protection and security arrangements Existence of effective back-up support Effectiveness of access control and security
	Location	Remoteness and/or co-location of facilities

Table III. Threat categories

Threat Categories	Entity	Nominal Weapon Aspirations				
		Number	Yield	Reliability	Delivery	Stockpile
1	Subnational	1 or 2	Any	Any	Truck/boat	No
2		5 to 10	Any	Any	Truck/boat	No
3	Non-Industrialized State	1 or 2	Any to 20 kt	50 - 95	Plane	Maybe
4		5 to 10	Any to 20 kt	50 - 95	Plane	Maybe
5		10 to 50	Any to 20 kt	50 - 95	Plane	Maybe
6	Developed State	1 or 2	Any to 20 kt	50 - 95	Plane	No
7		5 to 10	Any to 20 kt	95	Plane	Yes
8		10 to 50	20 to 200 kt	95	Missile	Yes

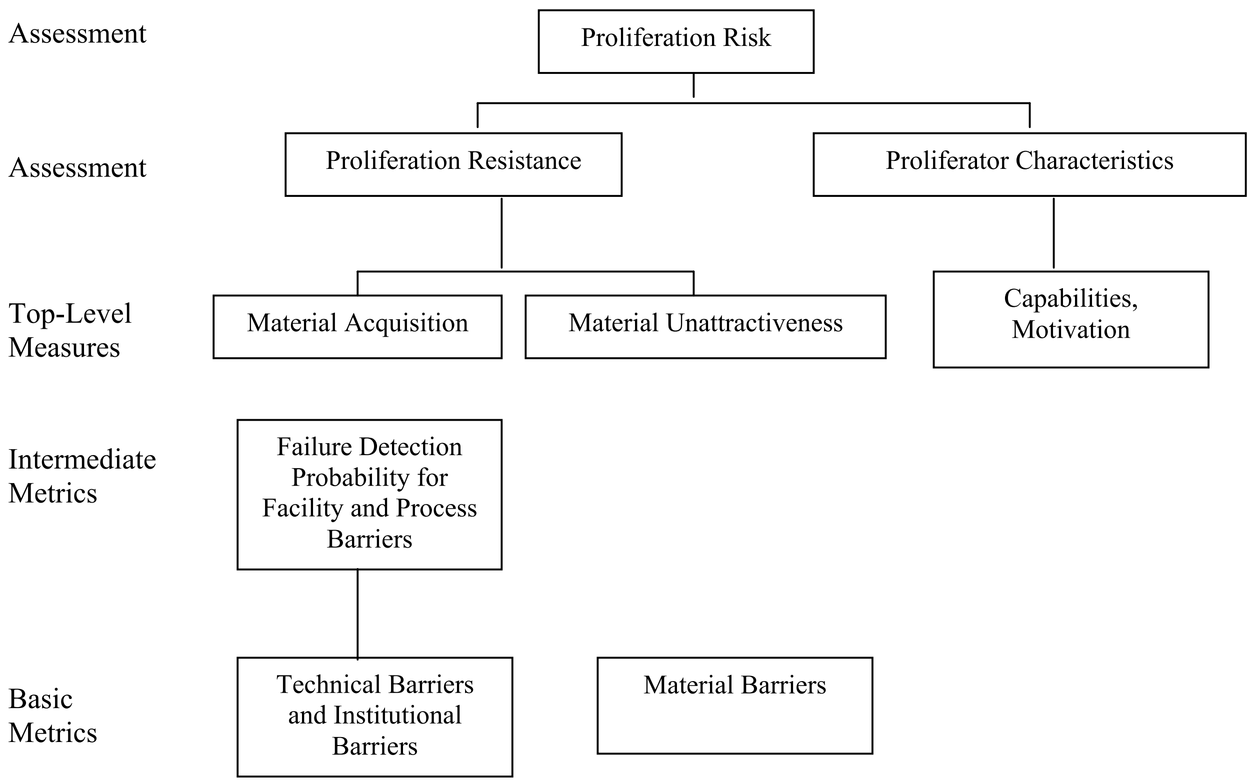

FIG. 4. Hierarchy of metrics.

8. SYSTEM SEGMENTATION

A nonproliferation issue relates to some type of system composed of facilities, processes, and controls. Frequently the system is an element or multiple elements of a fuel cycle system (for example, the element could be an enrichment facility). It is general practice to subdivide the system into discrete segments. The subdivision often occurs at the facility level, as illustrated in Figure 5. However, depending on the detail of the analysis approach, it may be necessary to further subdivide these facilities to the level of a distinct process line. For example, in the example in Figure 5, within the nuclear power plant, the accessibility and characteristics of fuel are different in the fresh fuel storage area, reactor core, and spent fuel storage pool. Thus, the nuclear power plant is subdivided into three elements. Similarly, at the ultimate storage facility, accessibility of material is different in the surface facilities than in the subsurface facilities. Once again, this facility is subdivided into two subunits for analysis. In contrast, the facilities at the front end of the fuel cycle involve only natural uranium, which is not a key target of proliferation. A number of facilities have been aggregated. Transportation between facilities can also be a point of diversion. Important transportation links can be identified as segments of the fuel cycle system in the same manner as facilities.

Typically, the analyst will compare the proliferation resistance of Fuel Cycle A for Threat 1 with the proliferation resistance of Fuel Cycle B for Threat 1 and similarly the resistance of the Fuel Cycle A for Threat 2 with the resistance of Fuel Cycle B for Threat 2, as illustrated in Figure 6. The analyst examines the proliferation resistance of each segment of the fuel cycle separately for each threat.

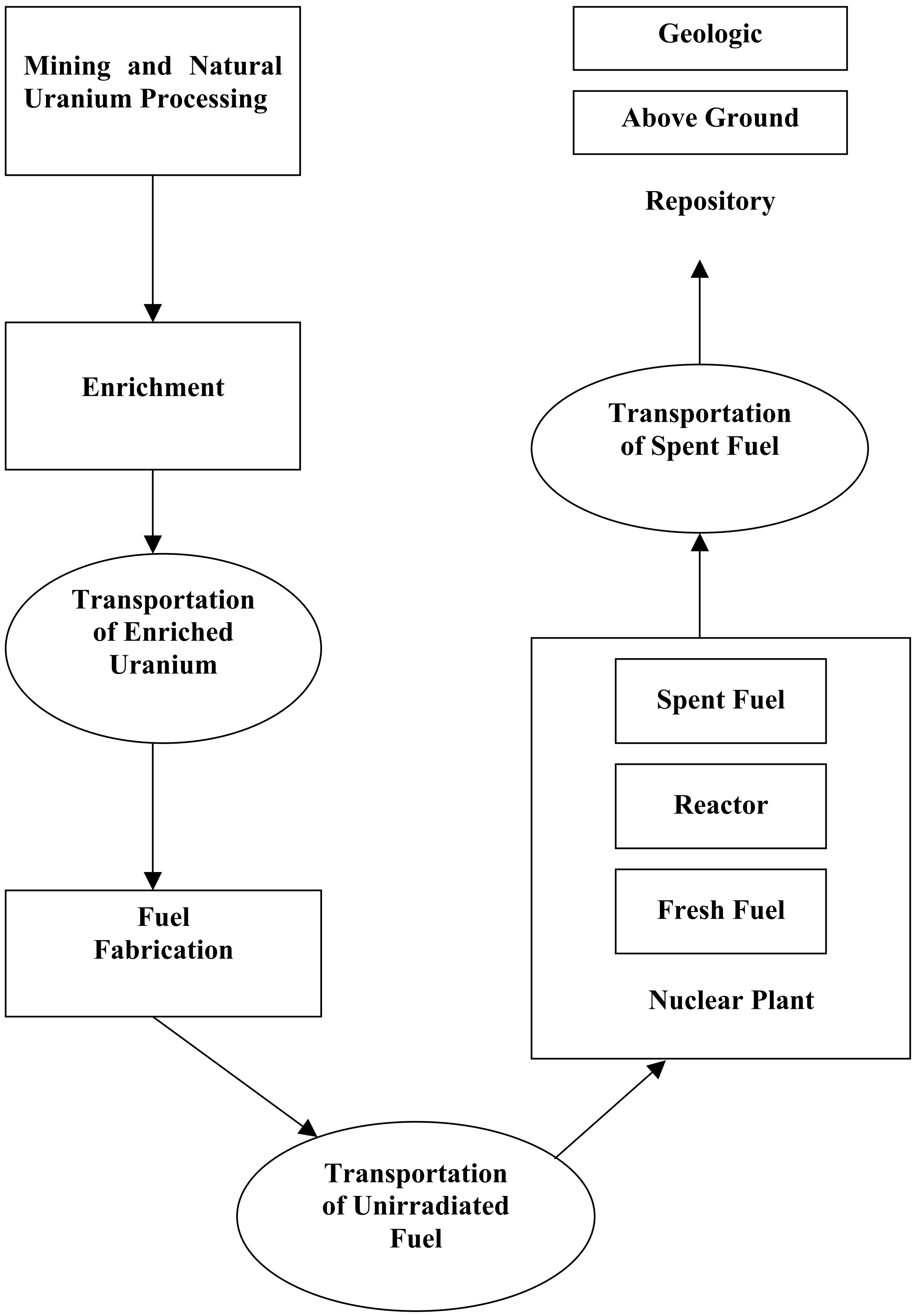

FIG. 5. Segmentation of fuel cycle.

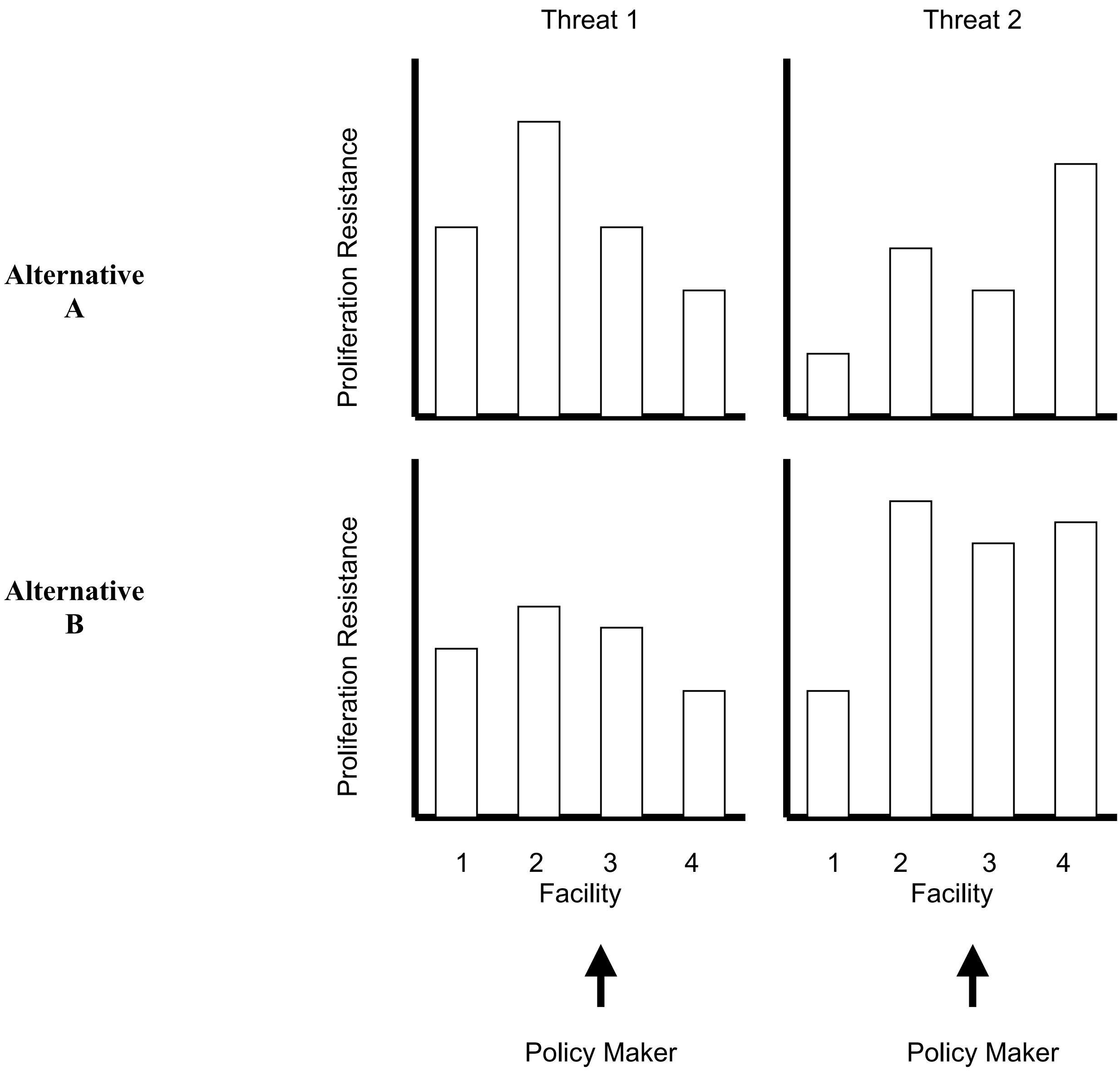

FIG. 6. Comparison of alternatives.

9. AGGREGATION AND PRESENTATION OF RESULTS

Figure 7 illustrates how the analysis is typically performed from the decomposed elements. The down and up arrows indicate an iteration over each of the elements in the top box. Thus, the complete analysis is performed for the first alternative and then for the second alternative. Within the analysis for each alternative, an assessment is made for each threat. Within each alternative and threat, an assessment is made for each segment of the fuel cycle (facility). The dashed box indicates the proliferation measurement algorithm, either a scenario analysis or an attribute analysis. For each alternative/threat/facility, the assessment measures the applicable barriers to proliferation. The weighting of metrics may occur within the measurement algorithm, or it may occur, as indicated in the figure, as a weighting of high-level metrics before the comparison of alternatives.

Two integrated methodologies are described in appendices to the guidelines document. The Risk-Informed Proliferation Analysis Methodology (RIPA) [4] is a scenario analysis approach that uses influence diagrams. The TOPS Barrier Analysis Method [5] is an attribute analysis approach. Both of these integrated methodologies are only in the development stage. Recommendations are made in this report for application studies in which these types of integrated methodologies would be further developed and evaluated. The aggregation of results and display of results are areas that are particularly in need of development.

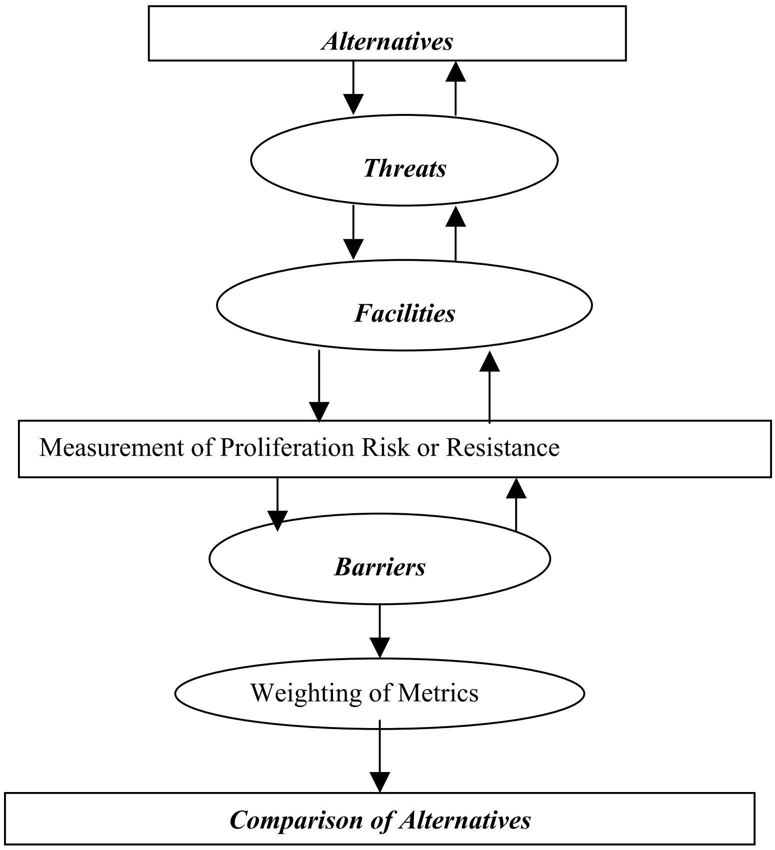

FIG. 7. Typical problem decomposition and analysis flow.

10. CONCLUSIONS

The guidelines document has advanced the process of developing integrated methodologies to address nonproliferation issues. Building upon previous work, the document takes the next step towards achieving a hierarchy of methodologies that can be employed with confidence, and that will be credible to a wide range of nonproliferation analysts and policy makers. A peer review of this work has been conducted by a group of experts in the area of nonproliferation assessment methods. The overall impact of this peer review was to validate the approach taken in the study and to assure the completeness of the methods surveyed and recommended.

Nonproliferation problems tend to be complex, and have defied the development of a universally agreed upon assessment methodology. However, there have been a number of attempts to apply rigorous methodologies to proliferation assessment problems. A number of promising techniques were identified during this development project. These analytical tools, often developed for analysis of other problems, can potentially be applied to nonproliferation assessments. Many of these tools will also be helpful in the development of integrated methodologies.

The study identified three general categories of analytical approaches with excellent potential for use in nonproliferation studies: attribute analysis, scenario analysis, and two-sided methods. Attribute analyses evaluate the effectiveness of barriers to proliferation, scenario analyses assess the pathways through those barriers, and the two-sided approaches explore the human interplay between adversaries. The three categories of approach are complementary in addressing the spectrum of nonproliferation issues that may be examined by NNSA and others.

Scenario analysis and attribute analysis methods can be used as the basis for integrated analysis approaches to the evaluation of the proliferation resistance or proliferation risk of nuclear systems. Two-sided methods, in particular wargames, can be used to examine proliferation issues that involve the interplay of opponents with opposing objectives. Each of these general approaches can employ one or more of the analytical tools identified in the study. For example, the scenario analysis and attribute analysis methods could each employ an analytical hierarchy process to systematically incorporate expert opinion into the weighting of metrics.

No fully mature integrated methods are available to assess proliferation risk or proliferation resistance. Before any integrated method can be used routinely, it will have to be tested and further developed through the performance of application studies. Scenario analysis and attribute analysis approaches have different strengths and weaknesses, which make one or the other preferable for a given application. The two methods can also be used effectively together.

Two very promising integrated methods, an attribute-based approach and a scenario-based approach, are described in this report. The Multi-Attribute Utility Barrier Analysis method and the Risk-Informed Proliferation Analysis methods have been partially developed but would require additional development effort to be used in a study. Although wargames are widely used in other fields, there are only limited examples of their use in examining nonproliferation issues.

A significant challenge to the development and application of integrated assessment methodologies is the selection of appropriate metrics. This document presents a hierarchy of metrics that can be used to convey the results of the assessment. There is a great deal of information produced in a nonproliferation study that must be presented to the policy maker in a manner that can be properly interpreted. Some aggregation of results must be made in the analysis to make the results interpretable. This presents a challenge for any of the methods that have been surveyed. The aggregation of metrics must be done in a manner that avoids loss of information, minimizes interdependencies, is traceable and provides useable information to the policy-makers. Detailed results should be documented to enable the policy maker to be able to trace higher-level results back to their lower-level causes.

NNSA, in collaboration with the Office of Nuclear Energy, Science and Technology of the Department of Energy, is currently applying the guidelines developed in this study to the formulation of an evaluation methodology for proliferation resistance of Generation IV nuclear energy systems.

11. ACKNOWLEDGEMENTS

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12. REFERENCES

[1] NNSA, "Guidelines for the Performance of Nonproliferation Assessments," (draft) November 2002, (final) mid-2003.

[2] PERLA, PETER, *The Art of Wargaming*, Naval Institute Press, 1990.

[3] SAATY, T.L., The Analytic Hierarchy Process, McGraw-Hill, ISBN 0-07-054371-2, 1980.

[4] ROCHAU, G.E. et al, "Risk-Informed Proliferation Analysis," Sandia National Laboratories Report, SAND2001-2020, June, 2002.

[5] NERAC, TOPS Task Force, "Annex: Attributes of Proliferation Resistance for Civilian Nuclear Power Systems," USDOE, October 2000. Available at: *http://www.nuclear.gov/nerac/FinalTOPSRptAnnex.pdf*

SIGMA: THE FIRST INTRINSIC PROLIFERATION RESISTANT URANIUM ENRICHMENT TECHNOLOGY

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Abstract. The future of the enrichment market faces a dilemma in using the currently available technologies. The new generation of enrichment plants should be economical, modular, with low investment rates, and low energy consumption, but also must be proliferation-resistant with industrial maturity. The SIGMA concept, an evolution of the GDT with some innovative components, overcomes all this requirements and it becomes the unique overall solution of the enrichment market dilemma, addressing for the first time the idea about a inherent proliferation resistant front end fuel cycle.

1. INTRODUCTION

1.1. Overview

As long as the main isotopic component found in the natural Uranium is the U^{238} (a non-fissile isotope), the natural Uranium must be processed in order to be used as nuclear fuel, enriching its contain of U^{235} (the fissile isotope). The Uranium could be processed to different enrichment levels, for being used in different kind of processes.

The Enriched Uranium with more than 2% of U^{235} could be used in a Nuclear Power Plant (NPP), moderated and cooled by Light Water (LWR). The LWR Technology is the world dominant Nuclear Reactor design, for nuclear electric power plants (NPPs).

The Heavy Water Reactors (HWR) constitutes the other main Reactor Technology. Even though the HWR were designed to be fueled by Natural Uranium, it is a fact that the use of Slightly Enriched Uranium (SEU) improves the economical behavior of this kind of NPPs.

Hence, the Uranium Enrichment Technology represents the basis of the civil nuclear industry, as the Enriched Uranium is the primary fuel used virtually in almost every nuclear reactor constructed in the world.

But the Enriched Uranium it is also used with military proposes, and historically this has been the driving force for the first Enrichment technology development.

1.2. Uranium enrichment technologies

The first significant development for Uranium Enrichment was made by the USA, in the Manhattan Project. The method was the Mass Spectrograph Isotope Separation Technology, based in an electromagnetic principle. This technique is simple and has a great separation factor, but uses a huge amount of expensive materials and electricity.

The Gaseous Diffusion Technology (GDT) was also developed within the frame of the Manhattan Project, surviving until our days. The GDT is the historical dominant technology in the Uranium Enrichment field, having produced more than 70 percent of the Enriched Uranium in the whole Nuclear Industry history.

Other that has reached industrial maturity is the Ultra Centrifugation Technology (UCT). It was developed at first by the former Soviet Union, in the early 50's. This method represents the actual successful competitor for the GDT.

The UCT is less expensive at a lower scale of production. This economical behavior is a competitive advantage, especially in a non-expanding market condition. Even though, the UCT has a Separative Work Unit (SWU) cost distribution with a great component of capital investment.

This technological status of the enrichment market leads in a world scenario where the actual production plants are in its end of life, and none of the technologies with industrial maturity reaches the economical and technical features required in this open and competitive world frame.

The replacement of the Enrichment plants will imply great investments in the next two decades, and the UCT are actually being taken as the least bad option for the actual plants replacement.

2. SWU FUTURE MARKET AND ACCEPTANCE CRITERIA

2.1. Future demand and challenges

Nowadays, a surplus offer of enrichment services is available over the world, but a great percentage of the installed plants are just in its last operation years. A more competitive market will face new challenges to follow the demand.

It can be estimated that the actual Gaseous Diffusion Plants (GDPs) will be out of service in this decade, beginning with the Americans, and ending with the last Europeans gaseous diffusion plants (2015). That leads in a fall of the enrichment services offer, which cannot be mitigated with the blending of the HEU from the disarmed nuclear devices, for the medium and long term.

Figure 1 shows the projected development of the Uranium enrichment services offer and demand, according to a study for medium and lower scenarios made by IAEA [1].

Comparing this situation with the Enriched Uranium demand, it can be see that the growing demand will overcome the offer between the years 2008 and 2010 by out of service of the aged plants.

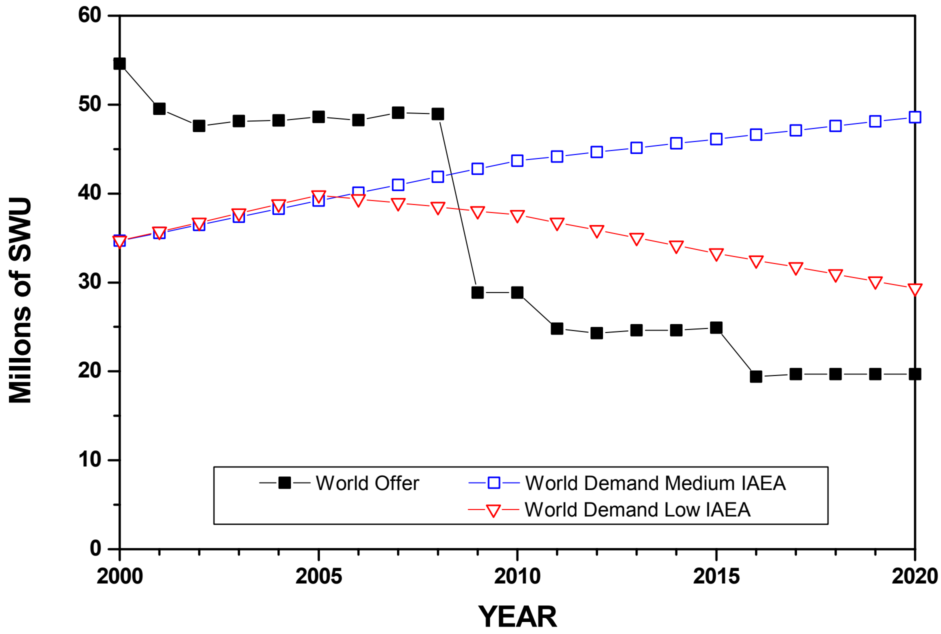

FIG. 1. Projected offer and demand of SWU for the next decades.

For the medium demand scenario it is not projected a significant increment in the SWU demand in the developed countries and Eastern Europe countries for the period 2000-2020, and for the lower scenario it will decrease one third of current situation. On the other way, a great grown appears in the developing country demand in both cases (China, Asiatic Southeast, Latin America, South and North Africa and Middle East).

Analyzing these two scenarios with the projected offer reduction, it is necessary to install 1 or 2 Millions of SWU average per year since 2008 to 2020.

At 1,000 USD/SWU as capital investment, UCT will requires 1,000 to 2,000 MUSD/year in new enrichment plants. This is far a way to be paid from the present cash flow of already active players in enrichment market (total sales of enrichment services is 4,000 MUSD/year). A fraction of the capital required could be directly or indirectly supplied by the governments if its perceived as "strategic", but only a fraction in present "deregulated" environment. A major fraction, will be supplied by the capital market at market discount rate. On this situation enrichment cost will increase up to 140 USD/SWU in order to recover the investment and improve the cash flow. This simply market feedback will potentially increase the cost of present OT LWR fuel cycle 1.5 mills/KWh, jeopardizing the competitiveness for all the fuel cycle because competitiveness margin with fossils sources is not high enough to remain competitive in all countries (effectively 1.5 mills/KWh is like to add 100 USD/KWe to a NPP capital).

Nowadays, almost all the enrichment services are located in developed countries, and have been also used for military purposes in many countries, given to a price structure that do not reflect the capital used to build this structure. So in a new competitive frame is logically possible that a future will produce new crisis.

Additionally to this situation, the central countries has constantly pressured to block the access of the developing countries to critical technologies, including the Uranium enrichment.

In the future, with the demand focused in developing countries, a serious problem will appear in central countries: Where will be the next generation of Enrichment plants? And who will pay the installations cost?

In this scenario, the market forces will interfere with the political constrains, related with the non-proliferation issues and commercial interests.

Even worst, the designers, trying to improve the economics for the new generation of enrichment technologies, looks to increase the separative factor and reduce the Uranium hold up, increasing the potential proliferation risk of all these new future technologies [5, 6].

3. ANALISYS OF THE ACTUAL ENRICHMENT TECHNOLOGIES

3.1. Laser technologies

There are two different methods comprehended within the Laser Isotope Separation (LIS) Technologies: The Atomic Vapor-Laser and the Molecular-Laser technologies.

In the first one, metallic Uranium is melted and vaporized into conditions of a rarefied atomic gas, and then is excited with a laser beam. This process leads to a differential photo ionization, that can be used to separate the isotopes using electromagnetic methods [2]. This kind of methods have been tested since the early 70´s, all over the world, but the most important developments were carried out in the US, within the AVLIS project frame and in France in the SILVA project. The Lasers used in AVLIS must be both tunable and powerful. To obtain that, dye laser pumped by copper lasers are used. The resulting laser beam is afterward improved in several amplifier stages. That leads to quite complex optical equipment that must be operated under industrial environment and scale. Moreover, handling of metallic Uranium vapor increases the complexity of the systems and sub-systems.

In the Molecular-Laser Technologies (MLIS), the excitation is produced to an entire molecule that contains a Uranium atom. Different molecules have been proved in MLIS methods, but even when each molecule leads to a characteristic process, all the MLIS techniques have the same scaling-up problem. Even when some problems of the AVLIS process have been solved in the MLIS technologies - like handling of metallic Uranium - some new troubles arise, especially those related with the required Laser power.

In all the LIS methods, the electrical consumption of the process is quite low, perhaps a 5% of the gaseous diffusion.

Even though the separation gain of the laser process is very high, what means just a single stage to reach commercial enrichments, the process must be operated at a very low pressure to achieve the differential atomic excitation, and so the necessary irradiation volume becomes huge.

This is the main problem of all the LIS methods. As long as the irradiation volume increases, the cost-related parameters of the LIS plants also grows-up. At the end, the need of a huge investment to build a high-tech optical system compensates all the good features of the technology, what leads in pretty attractive but economically unrealistic method.

In term of economics, the problem is that with the actual SWU price, it is impossible to pay the constructions of a LIS plant, even considering a negligible operation and maintenance (O&M) cost and using a near zero discount rate to evaluate the project.

A new M-LIS method has been under research by an US-Australian group: the SILEX technology. Even when there is not public data about the project, it is in the public domain that there is no clear evidence, up to now, that this technique could overcome the inherent economical problems of the LIS technologies.

This inherent economical problem could be best understood with a simple example: a standard commercial 4.5 MSWUs plant must process approximately 7,150 tons/year of natural uranium. That means the irradiation of around one ton of material per hour in conditions of rarified gas, in case of using the atomic vapor technology, or more in a MLIS method (depending on the type of molecule used). Rarified gas signify almost vacuum, so the volumes needed for irradiation are huge for a practical flow velocity in the irradiation chamber and its irradiation volume, and the capital costs are closely related with these volumes. Moreover, all the lab scale experiments are made in productions of micrograms per hour, so the scaling-up from lab to production scale is enormous, and so the uncertainties in the scaling-up methods.

3.2. Ultra centrifuges

In the Ultra Centrifuge Technology (UCT), the gaseous Uranium Hexafluoride (UF_6) is placed in a high-velocity rotating vessel. The centrifuge force acts separating the light and heavy isotopes, and using a counter-current velocity profile this separation is amplified. As long as the rotation creates a potential force gradient the only energy consumption is the irreversible friction loss of the rotating system and the electrical losses of the driven system, so the energy consumption of this method is quite low [3].

The enrichment gain of the UCT method strongly depends on the peripheral velocity of the rotating machine, and this characteristic is conditioned by the material resistance, or by the ultimate tensile stress over density (UTS/ρ) rate of the desired material. The difference could be significant, and different advanced materials could multiply the enrichment gain of the UCT process, just incrementing the rotating velocity. Due to this possibility of increment in the separation, great efforts are focused on the R&D of new materials and UC construction engineering. The high velocities related with the process imply significant complexity in the manufacture of the centrifuge rotor and especially in the bearings, which must withstand big forces due to slightly unbalanced rotors. The characteristic peripheral velocities of the UC machines goes from 400 m/sec using Aluminum alloys, to 1100 m/sec in Kevlar configurations. Anyway, the actual commercial machines could not achieve more than 600 m/sec. With the newest UC, just a few steps in cascade are needed to achieve PWR commercial level assays.

The SWU depends so much on the separation factor as on the flow of the machines. And this is the main problem of the UC method, because even when it could be achieved high enrichment with a small number of machines in series, each machine could manage just a very tiny amount of uranium for the limit to not reach the pressure de-sublimation. To generate a significant production rate of enriched uranium, a huge amount of machines in parallel are needed. Moreover, the SWU per machine is linear with the length of the rotor, and as the rotor becomes higher, the complexity of the construction and operation increases. Different strategies have been used, related with the optimal length of the machines: while some designers tries to obtain the longest machines, other design lines tries to find an optimal length. This optimal length depends on the cost-benefit rate of tolerance in rotor fabrication.

The very complex machine that results from the high-velocity requirements must be built in series of tens of thousands, to achieve a commercial production capacity. Construction tolerances are very strict in order to obtain a machine that could work for years, without maintenance, at almost tenth's thousand of RPMs. The bizarre conditions required for the construction of the machines results in a tremendously high capital cost for the machines.

The high capital cost of the UC machines prevails over the low electrical consumption of the technology, especially when the worldwide discount rates used for the nuclear projects evaluations monotonously grows-up. This is a significant problem in capital-intensive technologies that must be evaluated with low discount rates, and in long temporal periods, to enter into economical competition.

From the proliferation point of view, the UC method is potentially very risky because of the real possibility of high-grade sensible material diversion, and the difficult technical control. This is not a problem for Nuclear-Weapons-Owners countries, but it will be an international concern in case of being adopted by other nations.

3.3. Gaseous diffusion

The GDT stands on the statistical process of Uranium gas passing trough a porous membrane. The gas used for this process is the Uranium Hexafluoride (UF_6), because of its thermo-physical properties and the existence of only one isotope of Fluor.

Even though the GDT is not a complex technology, some main issues are not so easy to develop. The mayor technological challenge of the GDT is the development of the porous membranes.

The corrosiveness and toxicity of the gas used generates some others technological problems to be also solved: materials, seals, security requirements, compressors, etc.

As the enrichment factor of the GDT is very low (only 0,003 enrichment gain is the practical limit for each stage), is necessary to amplify the separative effect by connecting the diffusion stages in a cascade configuration. The GDP (Gaseous Diffusion Plants) is then made-up of thousands of stages, each one having a diffusion unit (set of membranes), one or two compressors, and a heat exchanger.

The construction of the GDPs started during World War II to produce enriched uranium for defense purposes. These plants were used primarily for this purpose through 1964. From 1959 through 1968, uranium enrichment production shifted primarily to supply the nuclear power industry. Nowadays, the GDPs produce the most part of the enriched uranium in the world.

The economic behavior of the GDT is mainly produced by the component scale economy as compressor efficiency decrease at low flow stages. A GD cascade has thousands of stages in series in which higher enrichments have lower mass flow. The smallest flow stages are the most expensive per SWU (higher component costs with less compressor efficiency). Thus a GDP will be competitive when the smallest stages are competitive, and this usually happens in GDP cascade configuration for at least a 3 MSWU/year plant capacity.

This behavior could be seen as a cascade paradigm, because is generated for the cascade concept itself, with small dependence on technology.

The Nusselt regime in the membranes and the ideal compression work produce an inherent energy consumption limit per SWU, with a value close to 2000 KWh/SWU. At 30 mills/KWh as a low energy price, the energy consumption cost is 60 USD/SWU. At a present market price of 100 USD/SWU the margin for capital amortization, operation, maintenance and profit is only 40 USD/SWU. Its clear than this margin is far to be enough for any new GDT plant, and is only produce by energy conservation in the more ideal case.

Then new GDP are far away to be competitive for price, energy consumption and scale economy, and then is clear wy UCT in the only present alterntive, with the limits and problems already shown.

4. SIGMA TECHNOLOGY

4.1. Introduction to the SIGMA technology

The gaseous diffusion technology developed by Argentina in earliest 80's [4] operated technically successfully, but nowadays it cannot compete with present SWU prices.

The GDT presents a robust and simple operation, but it only reaches competitive prices at huge industrial sizes and relative-high energy consumption.

As was explained, this behavior is related with the cascade paradigm of the GDT: all the improvements that could be done in the classical GDT are marginal.

The next step of Argentina's GD project is related with a new and revolutionary method for Uranium enrichment with gaseous diffusion. This method could reach, and it could even surpass, all the necessary requirements to be considered as the next generation of enrichment plants over the world.

The **SIGMA** (**S**eparación **I**sotópica **G**aseosa por **M**étodos **A**vanzados) concept represents the more drastic evolution of the GDT, adapted to the competitive world frame of the enrichment Uranium future market.

The SIGMA concept is focused in the reduction of the capital cost and the energy consumption, giving as a result, a small competitive production scale.

Some technical goals of the SIGMA concept are listed below:

— Reduced weight of main components.
— Low piping, joints and supporting materials.
— Low seal failure.
— Quick replacement of critical components.
— High compactness.
— High transportability of components.
— Reduced civil works.
— Modular design and construction.
— High overall energy efficiency.
— High separative factor by stage.

As long as the cascade paradigm of the GDT does not allow any significant changes in the plant economy, the SIGMA technology changed the cascade concept itself. This change opens a new design path and allows a new set of optimal design points, according with the restrictive requirements of future market.

The SIGMA concept includes a set of well-known components, which are used in an innovative way, and allows the capitalization of all the operation experience obtained in the classical GDT.

The improvements made are significant, increasing the performance of the components and systems by a factor 2 to 30.

All the innovations in the SIGMA concept are proved, representing a low technological risk. In the SIGMA concept, non proliferation is a central issue: the safeguards systems are included in the earliest design stages.

4.2. SIGMA design

Classical GDT use a stage, composed by porous membranes, compressor and coolers. To reach 5% of enrichment up to a thousand of stages (or more) are needed. This number of mechanical components and its related services produce the particular scale economy of the GDT. In Figure 2 could be seen the classical stages in series used in GDT.

In SIGMA design, steel vessel for the barriers are integrated in a single vessel, in with the different enrichment flows are separated with thin plates. Then the steel weight, flanges, elbows and fixing systems are strongly reduced. Up to 20 vessel are integrated in a single vessel.

In order to avoid connect up to 20 compressor to each SIGMA vessel (and then up to 20 bearings systems, motors, connections, etc.), a single compressor that compress up to 20 enrichment without mixing inside are used. This compressor called “multiflux compressor”, has been used for other technologies.

To recycle the depleted stream from the previous stage a gas gas injector is used avoiding use mechanical rotating devices.

The final SIGMA module scheme could be seen in Figure 3.

A SIGMA cascade, for enrichment up to 5%, could be build with close to 100 of SIGMA diffusers (instead of 1400 stages with classical GDT).

The final SIMGA configuration it’s slightly different to the Figure 3 scheme because for SIGMA is better to use double diffuser cascade instead of the single diffuser cascade as is usually used in GDT or UCT.

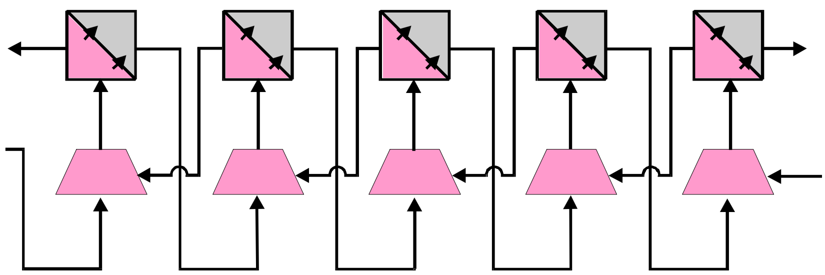

FIG. 2. Classical GDT stage configuration.

This double diffuser cascade enable to reach the double of separation factor per internal circulation stage, then up to 40 times more enrichment gain is obtained compared with GDT.

At the end less than 100 of modules could be designed to reach 5% or enrichment, all connected in serial cascade called SIGMA cascade, because the cascade connection of the double diffuser do not follows the classical double connection lay out.

To reduce the economic impact of the energy consumption, the SIGMA concept put in new perspective the problem. At 30 mills/KWh and 2000 KWh/SWU directly produce 60 USD/SWU in energy consumption without any doubt. But the electricity (highly sophisticated energy vector) is only used to produce the compression work, i.e. torque at a given rpm required by the compressor.

SIGMA reduce the order of magnitude of number of compressors (close or less than 100 in SIGMA, compared with 1400 in GDT). If an advanced Gas Turbine (GT) is directly connected to the compressor (replacing the cost of all the generator, transformers, transmission lines and electrical motors) only 35 USD/SWU is produced as ENERGY bill (not electricity) even considering the cost of the more expensive turbine compared with the electrical motors.

As the cost of transport uranium is negligible, then the enrichment plant could be directly moved to the gas field, the gas price is usually 30 to 50% lower, then the energy bill (if GT Compressor Driver are used), the total energy consumption cost is between 20 to 30 USD/SWU. At this point then this issue became only a variable cost that could be managed as other variable cost in other industries.

To use this approach, the SIGMA Module use and advanced GT directly coupled to the compressor, with capital, simplicity and efficiencies compatible with a 25 USD/SWU total energy cost.

As SIGMA is competitive even for a small 500,000 SWU/year enrichment output, the fuel requirement is really low in absolute terms, and then could be analyzed as a variable cost without significant logistic problems.

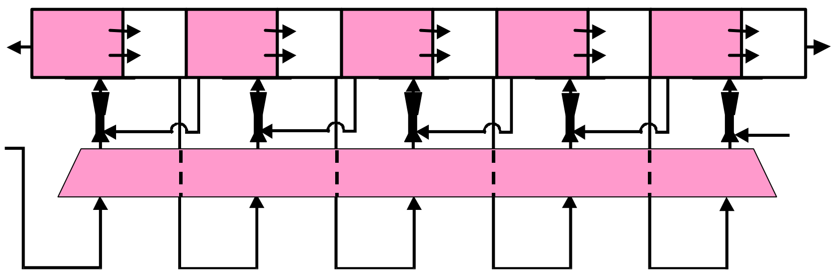

FIG. 3. SIGMA module configuration.

5. PROLIFERATION RESISTANCE

5.1. Present safeguards approach of enrichment plants

At present there are many proliferation resistance concept proposed for future advanced nuclear systems, many new reactors and new fuel cycles has been proposed [7]. As an example, only for LWR high burn up fuels, non fertile fuels or Thorium fuels has been proposed as future systems due to some inherent proliferation resistance characteristics of these alternatives [7].

But up to now there are no new proposals for present front end of nuclear fuel cycle. Recent studies usually assume that, in general, material present in the front end of LWR fuel cycle has very low attractiveness because it must be enriched significantly to be usable in weapon [8], but still there are concerns that plants designed to enrich LEU can be easily modified to produce HEU [8]. As was stated previously, it was a well-known issue that ultracentrifuge could be easily used to potentially produce direct usable weapon grade fissile material [5][6].

To survey a centrifuge plant, two key proliferation parameters are surveyed, the connections between centrifuge trains and the enrichment level itself [8]. Input and output quantities of uranium hexafluoride can be assayed very accurately, both for amount and for isotopic content. Give present uncertainties plants with up to 2 MSWU per year enrichment capacity appear safeguardable using current practice [9]. Since centrifuges operate at only a few torr, connections can be changed rapidly using plastic tubing. Swapped connectors can result in a plant yielding significant higher enrichment [8]. IAEA guards against changes in valve lineups or connections by surprise walk through inspections and surveillance cameras continuously monitoring passages between centrifuges. Part of the procedures involves safekeeping at the plant of original photographs of the cascade are, so that inspectors can compare them with the current layout, but has been addressed, the myriad of pipes, valves and connections inside a cascade, the visual acuity of the inspector can limit the utility of such comparison [9].

The principal difficulties of safeguarding gaseous diffusion enrichment plants involve the large amount of material normally present within the cascade, and the occurrence of some of the material plating out on the inside surfaces of pipes due to the small leakages [9], thus is clear that a gaseous diffusion producing LEU could be used to produce direct usable weapon grade fissile material. Then it's clear that gaseous diffusion have inherent proliferation resistant features compared with ultracentrifuges.

But the economical advantages of ultra centrifuges compared with classic gaseous diffusion are very well know and then at present there are gaseous diffusion plants that has been replaced, very recently, by new ultra centrifuges [10]. Looks like the market forces are selecting a technology because its cheaper, and then jeopardize the proliferation risk.

5.2. Safeguards cost

In practical term, sustainable energy development means that external factors (defining external as factor not including in present pricing values) should be considered along with traditional economic and technical issues in the planning and use of energy options [11]. Then economic development and proliferation resistant objectives should not be considered mutually exclusive but should be pursued as common and strongly linked goals.

Then, to evaluate if this market tendency its sustainable or not, the additional external cost of safeguards need to be calculated in order to compare with present SWU cost.

A first rough estimation could be obtained using published IAEA safeguards efforts for enrichment plants [12] and the IAEA safeguards budget [13]. A first looks shows that present IAEA enrichment safeguards efforts are a significant proportion of current total safeguards efforts, using close to 700 persons days of inspection effort, and if its considered that IAEA enrichment safeguard efforts are devoted mainly for URENCO and PNC ultracentrifuges plants [9], with a total capacity of 3.1 to 1.1 MSWU/year plant capacity respectively [14], the total safeguards cost could be close to 4 USD/SWU. This estimation has been done considering all the support structure and specific equipment required and its related R&D [15, 16], national authority safeguards costs and operators costs.

Even the order of magnitude of the safeguards efforts, when it's compared with present SWU market price, it's significant and couldn't be neglected without further consideration. As an example, in order to replace the older gaseous diffusion plants and cover the additional SWU demand increase by present nuclear energy growth forecast, in the year 2020, up to 20 or 30 MSWU new centrifuge plants will be needed, and at that time could need to be under safeguards. The additional inspector efforts need to be multiplied for a factor of 8, totaling an inspection effort of 6 thousand of inspection days/year, implying close to double the IAEA inspection efforts devoted to all nuclear energy power.

Even that's its clear that centrifuge savings compared with gaseous diffusion are bigger than the safeguards cost, the market its selecting a technology not in a holistic point of view, because the cheapest option is not the least proliferation resistant option too.

6. SIGMA PROLIFERATION RESISTANCE APPROACH AND PROJECT STATUS

6.1. SIGMA safeguard approach

In the SIGMA concept, non proliferation is a central issue: the safeguards systems are included in the design itself, using all the non proliferation advantages of the GDT, multiplied by some innovative features that allows the use of NDA techniques for **on-line hold-up control**.

The SIGMA technology inherits all the proliferation resistant features from Gaseous Diffusion Technology, and incorporates some new features that results in a near zero risk of proliferation.

At present oil well logging technology develop many remote NDA probes, that could be used to measures gamma and neutron radiation in compact and self contained detectors, and transfer all data to remote operators

As in the SIGMA module, many old GDT stages are in a single vessel, in the center of the vessel could be put a NDA probe to measure both gamma and neutron radiation in order to measure the uranium quantity and its enrichment.

As the number of modules are very low, and the probes NDA relative inexpensive, each module have its own detector, working continuously, and then the measurement time is not an issue because the diversion time are several orders of magnitude greater. This type of measurement are very useful for the operator (on line cheap enrichment measurement) and for safeguards, because in the SIGMA plants will be available for the inspector. This probe could be even combined with start of the art remote safeguards techniques in order to send the information on line with the safeguard authority in its headquarters.

As the number of stages is close to 100 connected in series, its practically impossible to use a SIGMA plant to produce HEU. The valves positions, pipes, etc. do not need to be checked at all because there is no reconfiguration risk in any way.

The diversion of large quantities of LEU, or feed/withdraw of undeclared material will be on-line detected by an unambiguous alarm of the on line safeguard NDA probe system. This is a radically new proliferation resistant characteristics of SIGMA compared with any other enrichment technology (Table I).

Table I. Comparison of proliferation risk of actual enrichment technologies. [5, 6]

Characteristic	**Ultra Centrifuge**	**LASER Methods**	**Gaseous Diffusion**	**SIGMA**
Characteristic Times	Rapids	Rapids	Slow	Slow
Re-Configuration Risk	High	Very High	Low	Very Low
Hold up	Zero	Zero	High	High
HEU diversion capacity	Medium	High	Low	Very Low
NDA techniques	Acceptable	Unknown	Acceptable	Excellent
On line analysis	No	No	No	Excellent
Overall Proliferation Risk	Medium	High	Moderated	Near Zero

6.2. *SIGMA project*

In SIGMA project there are several groups working in different areas of conventional gaseous diffusion technology, and in SIGMA specific areas.

A new generation of membranes has been developed. The new membranes have already reached industrial maturity. These membranes could be used so in conventional GDT as much as in a SIGMA module.

Design and economy groups are doing economical studies, in order to improve the main design characteristics of the SIGMA technology. Several numerical models have been developed since the project has begun. Nowadays, the design has a complete set of validated models in different areas:

— Process and SIGMA cascade calculations.
— Membrane economical design and diffusion calculations.
— SIGMA Plant overall calculation.
— Physical model and economic behavior of each main SIGMA component.

Experimental groups are also working in validation of various models, membrane development, safeguard NDA probe design, and in the SIGMA pilot-scale concept demonstration facility, which has been built in the Pilcaniyeu Technological Installation simulating UF_6 with inert gases (in order to performed changes in minutes in all pipes and connections).

The present pilot plant successfully produce the required data to validate the models and the design code.

6.3. *Concept demonstration experimental facility*

In order to study the design's feedbacks and some particular behaviors of the SIGMA technology, an experimental facility has been constructed, and is now being operated by a group of experimental engineers.

This facility is being used to prove the main innovative components incorporated to the design, as well as some minor engineering technical solutions.

An important set of parameters has been obtained as a result of the experiences with the concept demonstration facility, in order to improve the final commercial plant design with detailed engineering models, and to produce the engineering data for the next experimental step.

7. COMPARISON BETWEEN THE DIFFERENT ENRICHMENT TECHNOLOGIES

Keeping in mind the amount of time involved in design and construction of an enrichment plant, and the end of life of the actual plants, it is a fact that the new generation of enrichment plants should be chosen in the next few years.

The potential investor in a new enrichment facility will should make his decision based upon the economical, technical and political aspects of the available technologies and the market.

The economical aspects of the actual and future enrichment market were discussed earlier in this paper. The technical aspect changes continuously, and a new technological idea could appear in any moment, changing the entire situation.

In reference with the political aspects, it is unlikely that a major change occur, since the nuclear weapon situation is still a growing global threat.

Until now, the investor decision was unique. The only economically acceptable and technologically reasonable method for Uranium enrichment was the UCT.

Even when the Gaseous Diffusion Plants were considered so simple, reliable and well known, its prohibitive economic size makes this technology an unacceptable choice.

As the demand was shifting toward developing countries [1], the investor will should consider installing the plant in a non nuclear weapon owner country. Up today, making some punctual exceptions, this seems impossible, because of the implied proliferation risk.

To build large transnational companies to supply enrichment services worldwide has been recently propose to reduce the proliferation risk of enrichment [17]. This idea is far to be new, and has been proposed 30 years ago [18]. This idea was implemented in the seventies in one large enrichment services company, including a developing country in the capital rising with the corresponding shares distribution. The experience shows that this solution, transparent on the international forums, could be seen in the practice as an intention to transfer money to develop country industry in a self - protected bussiness with the more painful and expensive resource in developing country: capital for investment. The experience shows that this solution does not solve the problem of proliferation at all, even could put nuclear energy as an unreliable source for stable energy supply for developing countries.

If this next generation technology could offer a reliable and proliferation resistant method, perhaps what today seems impossible could be a reality tomorrow.

In the Tables II, a comparison of the economical, technical and proliferation risk aspects of the different enrichment technologies in shown.

Table II. Comparison between actual enrichment technologies.

Characteristics	Ultra Centrifuge	LASER Methods	Gaseous Diffusion	SIGMA
Operation Costs	High	Unknown	Moderate	Balanced
Energetic Costs	Low	Low	High	Balanced
Capital Costs	High	Unknown	Moderate	Balanced
Economical Size	Small	Unknown	Very Big	Low
Modular Design	No	No	No	Yes
Maturity	Commercial	Development	Commercial	Proved Changes
Future Expansion Potential	Good	Developing	Poor	Very Good

8. CONCLUSIONS

The future of the enrichment market outlines a without solution dilemma, using the up today available technologies.

Today's plants are in its end of life, and none of the existent technologies are capable of offering the complete solution to the presented problem.

The new generation of enrichment plants should be economical, modular, with low investment rates, and low energy consumption. But also must be proliferation-resistant, and off course should be reliable, robust and with industrial maturity.

The SIGMA concept, an evolution of the GDT with some innovative components, overcomes all this requirements and it becomes the unique overall solution of the enrichment market dilemma, addressing for the first time the idea about a inherent proliferation resistant front end fuel cycle.

9. REFERENCES

[1] WAGNER, H.F., FOSTER, J., et al., "Global Energy Outlook: Key Issue Paper Nº 1", Proceedings of the International Symposium held in Vienna, Austria, 3-6 June 1997, Nuclear Fuel Cycle and Reactor Strategies: Adjusting to New Realities. Ed. IAEA, Vienna, Austria, 1997: Annex 1: Estimation of Uranium and fuel cycle service requirements, page 58.

[2] PAISNER, J.A., Atomic Vapour Isotope Separation, Applied Physics B Vol. 46, pages 253-260 (1988)

[3] ALBRIGHT, D, BERKHOUT, F., WALKER, W., Plutonium and Highly Enriched Uranium 1996, World Inventories, Capabilities and Policies, SIPRI, Oxford University Press, (New York), 1997

[4] CASTRO MADERO, C.Y., TAKACS, E., Politica Nuclear Argentina: ¿Avance o retroceso?, Buenos Aires, Ed. El Ateneo, Capítulo III, pág. 26 (1991)

[5] SILVENNOINEN, P., VIRA, J., "Quantifying relative proliferation risks from nuclear fuel cycles" Progress in Nuclear Energy, **17**(3): 231-243 (1986).

[6] AHMED, S., HUSSEINY, A.A., "Risk Assessment of alternative proliferation routes", Nuclear Technology, **56**: 507:515 (1982).

[7] SHEA, T., Reactor Design / Fuel Cycle Principles Related to Proliferation Resistance. Workshop on Proliferation Resistance in Innovative Reactor Fuel Cycles. Como, Italy, 2-6 June 2001.

[8] Join Study on Enhancing the Proliferation Resistance of Nuclear Technologies. Joint Draft Text USDOE and MINATOM-RF. Proliferation Resistance Nuclear Technologies. March 24, 2000.

[9] Nuclear Safeguards and the International Atomic Energy Agency. Office of Technology Assessment. Congress of the United States. Washington D.C. 1995.

[10] JUNJIE LI, Progress of China's Nuclear Fuel Cycle Industry. 4th Meeting of the Technical Working Group on Nuclear Fuel Cycle Option. Vienna, 15 – 17 October 2001.

[11] Health and Environmental Impacts of Electricity Generation Systems: Procedures for Comparative Assessment. IAEA Technical Report Series No. 394. 1999.

[12] HURT, D., IAEA Safeguards at Nuclear Facilities. Workshop on Proliferation Resistance in Innovative Reactors and Fuel Cycles. Como, Italy, 2 – 6 July 2001.

[13] The IAEA Safeguards System. Ready for the 21st Century. International Atomic Energy Agency. IAEA, September 1997.

[14] Country Nuclear Fuel Cycle Profiles. IAEA Technical Report Series No. 404. 2001.
[15] BAKER, J.S., DEKKER, B.G., FRIEND, P.J., IDE, K., Developments in safeguards as applicable to URENCOS's enrichment plants. International Nuclear Safeguards 1994: Vision for the Future. Proceeding of a Symposium on International Safeguards. Vienna, 14 – 18 March 1994.
[16] OMAYE, M., SUZUKI, S., YUZAWA, S., NOTAGASHIRA, T., Outline of the Safeguards Implemented at the Rokkasho Uranium Enrichment Plant. International Nuclear Safeguards 1994: Vision for the Future. Proceeding of a Symposium on International Safeguards. Vienna, 14 – 18 March 1994.
[17] Guidance for the evaluation of innovative nuclear reactors and fuel cycles. IAEA Tecdoc 1392. International Atomic Energy Agency, June 2003.
[18] INFCE. International Nuclear Fuel Cycle Evaluation. 27 – 29 November 1978. INFCE Volume 2. Enrichment Availability, International Atomic Energy Agency 1980.

INNOVATIVE NUCLEAR FUEL CYCLE TECHNOLOGIES MEETING USERS REQUIREMENT IN THE AREAS OF ECONOMY, SAFETY, ENVIRONMENT AND WASTE, PROLIFERATION-RESISTANCE AND CROSS-CUTTING ISSUES

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Abstract. The International Atomic Energy Agency (IAEA) had organized a Technical Specialists Meeting (TSM) at Vienna during April 2-4, 2003 to discuss 'Innovative Fuel Cycle Technologies' as part of the Agency's "Innovative Nuclear Reactors and Fuel Cycles Programme" (INPRO). Some 36 experts from Argentina, Belgium, Canada, China, France, Germany, India, Japan, Republic of Korea, Russian Federation, European Union, World Nuclear Association and IAEA had participated in the meeting. In all, 21 Invited Papers were presented in 4 technical sessions covering topics related to: (i) Front-end of Fuel Cycle, (ii) Energy Conversion, (iii) Back-end of Fuel Cycle and (iv) Innovative Fuel Cycle Concepts. In the Panel Discussion and Concluding Session, thrust areas were identified and a few recommendations were made for consideration of IAEA.

1. INTRODUCTION

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) of IAEA was conceived in the General Conference of September 2000. In subsequent General Conferences of 2001 and 2002, Member-States reinforced their support for such activities. Additional endorsement came in 2001 and 2002 through the UN General Assembly resolutions, which emphasized "the unique role that the Agency can play in developing user requirements and in addressing safeguards, safety and environmental questions for innovative reactors and their fuel cycles the need for international collaboration in the development of innovative nuclear technology". INPRO is an Agency-wide Project, currently involving 13 Member-States (MS), namely Argentina, Brazil, Canada, China, Germany, India, Netherlands, Republic of Korea, Russian Federation, Spain, Switzerland, Turkey and the European Commission (EC).

In July 2002, the United States initiated the Generation IV International Forum (GIF) with the aim of developing advanced nuclear energy systems for international deployment beyond the year 2030. Presently, 10 countries, namely Argentina, Brazil, Canada, France, Japan, Republic of Korea, Republic of South Africa, Switzerland, the United Kingdom and the United States are participating in GIF. Six reactor concepts have been selected for international collaborative R&D and for each concept a lead country has been identified. The reactor concepts and the lead countries are: (i) Gas-cooled Fast Reactor (USA), (ii) Lead-cooled Fast Reactor (Switzerland), (iii) Sodium-cooled Fast Reactor (Japan), (iv) Supercritical Water-cooled Reactor (Canada), (v) Very High Temperature Reactor (France) (vi) Molten-Salt Reactor (lead country yet to be decided). There is lot of commonality in the overall approach and objectives of INPRO and GIF. Both programmes aim to promote international co-operation to ensure sustainable development and economic competitiveness of nuclear energy and to meet high standards in the area of safety and reliability of nuclear power plants, environment, waste management and proliferation-resistance. As part of IAEA's activities on INPRO, a Technical Specialists Meeting (TSM) was organized at Vienna during April 2-4, 2003 to enable Member-States organizations, stakeholders and policy makers to exchange reliable state-of-the-art information on current and future trends in 'nuclear fuels cycle technologies'. In the TSM, some 36 experts from Argentina, Belgium, Canada, China, France, Germany, India, Japan, Republic of Korea, Russian Federation, European Union (EU), World Nuclear Association (WNA) and IAEA had participated. In all, 21 invited papers were presented in 4 technical sessions. In the concluding session, a panel of experts identified and recommended a few thrust areas for consideration of IAEA.

The TSM brought out the extraordinary diversity of fuel cycle options in Member-States – from once-through uranium fuel cycle involving relatively inexpensive low enriched uranium (LEU) technology to plutonium breeding and utilization in LWRs, PHWRs & FBRs – thorium fuel cycles – from direct disposal to partitioning and transmutation of nuclear wastes. In addition, the demand and supply positions of uranium, the uranium market & current price, the type of uranium mining operations, the major uranium producers and enrichment facilities all over the world were discussed. Further, the need for economic indicators and critical intercomparison between 'nuclear' and 'fossil' as primary sources of energy with that of renewable energy like 'wind' and 'solar' was also brought out. The present dissertation gives a list of the papers presented by experts in the TSM and summarises the status of uranium and thorium resources, the fuel cycle programme in several Member-States, the conclusions of the panel discussion and the recommendations to the IAEA.

2. FRONT-END OF FUEL CYCLE

(i) 'Mining, Milling and Enrichment'	: S. Kidd, WNA, London
(ii) 'SIGMA: Fuel Cycle Approach for INPRO'	: P. C. Florido, Argentina
(iii) 'Fabrication Technology of Uranium, Plutonium and Thorium bearing fuels for Nuclear Power Reactors'	: C. Ganguly, India

The known conventional resource of uranium is in the range of 4 million tons of which the reasonably assured resources (RAR), costing less than 130 US $/kg 'U', account for nearly 3 million tons. In addition, there are approximately 11.5 million tons of undiscovered resource. Further, large quantities of uranium are available from unconventional sources like phosphates and sea-water. Presently, some 37,000 tons uranium is produced annually all over the world of which 30% is produced in Canada, 22% in Australia, 19% in Africa, some 18% in CIS countries, 4% in USA, 3% in Europe and the balance in several countries. The demand of uranium is in the range of 65,000 tons per year. The gap between demand and production of uranium, approximately 45%, is currently met from secondary supplies, which include ex-military high enriched uranium (HEU), reprocessed uranium (RU), depleted uranium (DU) and civil and military plutonium (as substitute for uranium) in the form of mixed uranium plutonium oxide. The secondary supplies are likely to decline to 4-6% by the year 2025. The demand of uranium has not increased as projected in the 1980s, as a result of which the uranium price has remained stable and low (US $10-11/pound U_3O_8) in the international market. The uranium mining activities are being closed in France, Spain and Gabon and there is not much incentive to open new mines. More than 60% of the world's known recoverable deposits of uranium are located in Australia, Brazil, Kazakhstan, Uzbekistan and a few States in Africa, which have either no or very limited nuclear power programme. There is a need to create new uranium markets in countries, like India, with an ever-increasing and ambitious nuclear electricity programme. India is of the view that this is possible if the present international treaties and guidelines related to uranium trading are modified and made more practical. These documents should not only ensure non-proliferation of uranium and plutonium (obtained from uranium) for non-peaceful purpose but also facilitate transparent international trade and collaboration, for peaceful use of uranium and plutonium for generation of clean electricity, between large uranium producing countries with no or very little domestic nuclear power programme and countries which have ambitious nuclear electricity programme but is handicapped by not having access to uranium in the international market and has to depend on the high price of indigenous uranium because of very low grades and the need for deep underground mining operations. Recently, there has been some growth in uranium mining and milling activities in Canada, Australia, Kazakhstan and Uzbekistan. The mining activities are mostly carried out in big conventional open pit or underground mines (eg. McArthur River, Rossing, Olympic Dam, etc.) by large mining companies, namely Cameco, Cogema, Rio Tinto. However, in recent years, smaller but low cost 'in-situ leaching' (ISL) is gaining popularity, mainly in Kazakhstan. In the area of uranium enrichment, Gaseous Diffusion Technology (GDT) have so far been utilized for producing some 70% of enriched uranium for the nuclear industry. These plants are expensive and old and are being phased out. Ultra Centrifugation Technology (UCT) is becoming more and more popular because it is less energy intensive and economic. The Laser isotope separation technologies based on Atomic Vapour Laser Isotope Separation (AVLIS) and Molecular Laser Isotope Separation (MLIS) has the inherent advantage of 'selectivity' but are yet to be commercialized.

Thorium is 3 times more abundant than uranium in earth's crust. Monazite, a mixed thorium rare earths (RE) phosphate, has been exploited in the past mainly in Australia, Brazil, China, India, Malaysia, South Africa and USA. Presently, some 5,000 tons of monazite is produced and processed annually all over the world, mostly in India (for recovery of thorium and uranium) and China (for recovery of RE). The proven monazite reserves in India are in the range of 8 million tons, which occur, along with other heavy minerals, in the southern and south-eastern sea coasts. The monazite contains 8% ThO_2 and 0.35% U_3O_8.

Argentina is developing a simple, advanced and Integrated Pressurized light Water Reactor (IPWR) concept called CAREM – IPWR with electric power in the range of 150-300 MWe to meet the requirement of small and medium size developing countries. The CAREM reactor is inherently safe, works on natural convection, uses mini helicoidal steam generator and a novel self pressurized system and is highly economic with capital cost in the range of 1000 $/kWe. As a first step, a CAREM 25 IPWR is being planned and is presently at the detailed engineering stage. The CAREM reactor would be working on 'once-through' fuel cycle based on low enriched uranium. For this, a novel and low cost uranium enrichment technology, SIGMA (Separacion Isotopica Gasceosa por Metodos Avanzados) based on gas diffusion and use of much lesser number of compressors has been developed with the objective of reducing capital cost and energy consumption. Only 100 modules of serial SIGMA cascades could lead to 5% U^{235} enrichment, which is adequate for nuclear power reactors. Thus, SIGMA has built-in proliferation-resistant features because with such limited number of cascades, it is not possible to produce U^{235} enrichment beyond 20%. In addition, SIGMA also has novel safeguard features and on-line materials accounting system.

India has limited uranium but vast thorium resources. Accordingly, a three-stage indigenous nuclear power programme is being pursued, linking PHWR, LMFBR and Thorium-based Advanced Reactors, utilizing 'closed U^{238}-Pu^{239} & Th^{232}-U^{233} fuel cycles'. Presently, twelve units of PHWR 220 MWe (a few derated) and two units of BWR 160 MWe are under operation with total installed power of 2,720 MWe. Eight water-cooled reactors of total capacity 3,960 MWe are under construction, including four PHWR 220, two PHWR 540 and two VVER 1000 (with Russian collaboration). The Department of Atomic Energy has set a target of installing 20,000 MWe nuclear power by the year 2020, which would include PHWR 680, LMFBR 500 and LWR 1000. In India, PHWR fuel cycle has matured in all respects. Mining & milling of uranium, fabrication of UO_2 fuel, their large scale utilization in PHWR 220 units and reprocessing of spent fuel are being carried out on an industrial scale for more than two decades. In the Indian PHWRs, depleted UO_2 and ThO_2 bundles are being used for neutron flux flattening of initial cores during start-up. As a first step to LMFBR programme, India could leap-frog and use an advanced LMFBR fuel, namely a hitherto untried plutonium rich mixed uranium plutonium monocarbide, as driver fuel in Fast Breeder Test Reactor (FBTR). The mixed monocarbide fuel has achieved a burn-up of more than 100,000 MWd/t without any failure. ThO_2 assemblies are being used in FBTR as radial blanket material. Construction of a Prototype Fast Breeder Reactor of 500 MWe (PFBR 500), using mixed uranium plutonium oxide (MOX) with 20-25% PuO_2 as drivel fuel would start soon. Design and development are underway for an Advanced Heavy Water-cooled Reactor of 300 MWe (AHWR 300) as part of the third stage of the nuclear power programme. AHWR is a vertical pressure tube type, D_2O moderated, light boiling water-cooled reactor using $(Th,Pu)O_2$ and $(Th,U^{233})O_2$ (<4% fissile material) as fuel. A pilot plant has been commissioned for reprocessing of spent thoria assemblies for recovery of U^{233}. R&D activities are underway for nuclear desalination, Compact High Temperature Reactor (CHTR) and its fuel cycle and Accelerator Driven System (ADS). As part of fuel cycle programme, in recent years, there has been significant enhancement of uranium exploration and mining activities. A new underground uranium mine, the fourth in the series, has been opened and there has been significant capacity expansion in production of zirconium sponge, alloys and components and natural uranium oxide fuel bundles for the operating PHWR 220 and the forthcoming PHWR 540 and PHWR 220 units. A new monazite processing project, THRUST has been initiated for recovery of uranium as ADU and purification and conversion of thorium hydroxide to thorium oxalate for long-term storage.

Presently, the ‘powder-pellet’ route is universally followed for fabrication of UO_2 and $(U,Pu)O_2$ fuel pellets using UO_2 and PuO_2 powders as starting materials. The UO_2 powder is manufactured by Integrated Dry Route (IDR), Ammonium Di-Uranate (ADU) or Ammonium Uranium Carbonate (AUC) processes. The PuO_2 powder is obtained by controlled calcination of plutonium oxalate. Likewise, ThO_2, $(Th,Pu)O_2$ and $(Th,U)O_2$ pellets are also fabricated by ‘powder-pellet’ route. The ThO_2 powder is mostly obtained by calcination of thorium oxalate. The ‘powder-pellet’ route has also been utilized for fabrication of advanced LMFBR fuels, namely mixed uranium plutonium monocarbide and mononitride. First, monocarbide and mononitride powders are prepared by controlled carbothermic reduction of tableted mixture of UO_2, PuO_2 and C in vacuum and flowing nitrogen respectively followed by crushing and milling. Next, the monocarbide and mononitride powders are subjected to cold pelletisation followed by high temperature sintering in inert atmosphere to obtain monocarbide and mononitride fuel pellets. The ‘powder-pellet’ route involves generation and handling of fine powders and is associated with the problem of ‘radioactive dust hazard’. In most cases, the fine powders are not suitable for remote fabrication because they are not free-flowing and are required to be granulated to obtain suitable ‘press-feed’ material. A novel dust-free Sol-Gel Microsphere Pelletisation (SGMP) process has been developed for remote and automated fabrication of highly radioactive Pu and U^{233} bearing oxide, carbide and nitride fuel pellets using relatively coarse (500-1000 micron) and free-flowing sol-gel derived microspheres for direct pelletisation and sintering. The hydrated oxide or mixed oxide microspheres are prepared by ammonia internal or external gelation processes using nitrate solutions of uranium, plutonium or thorium as starting materials. For preparation of uranium plutonium monocarbide and mononitride microspheres, carbon black is added to the sol or solution prior to gelation. The SGMP process has been combined with Low Temperature oxidative Sintering (LTS) for fabrication of high density UO_2 and $(U,Pu)O_2$ fuel pellets. The SGMP-LTS route has been utilized for manufacturing and irradiation-testing of several UO_2 fuel bundles in PHWR 220. A few MOX fuel pin has also been manufactured by this process and successfully irradiated in research reactor. As part of AHWR fuel development activities, process flowsheets based on classical 'powder-pellet' route has been developed for $(Th,Pu)O_2$ and $(Th,U)O_2$ fuels. A few $(Th,Pu)O_2$ test fuel pins were fabricated earlier and successfully irradiated in CIRUS reactor. For $(Th,U^{233})O_2$ fuel SGMP and 'pellet impregnation' processes have been developed and demonstrated.

3. ENERGY CONVERSION

(iii) "Advanced fuels for Plutonium Management in PWRs" by A.Vasile, Ph. Dufour, H, Golfier, J.P. Grouiller, J.L. Guillet, Ch. Poinot, G. Youinou, A.Zaetta.
(iv) "Advanced Fuel Cycles in the ACR: Meeting INPRO User Requirements" by G.R.Dyck and. P.G. Boczar.
(v) "Fuel Cycles for HTRs" by Dominique Greneche, France
(vi) "Basic Requirements for Innovative Nuclear Technology of large scale Nuclear Power System and their implementation through BREST Concept" by Yu S. Cherepnin, Russia
(vii) "Innovative Technologies of reprocessing and refabrication of nuclear fuel" by G.Uchiama, H. Ojima, T. Koyama, M. Myochin and Y. Kihara, Japan.
(viii) "Innovative Nuclear Energy Systems for Inherently Protected Plutonium Production" by M.Saito, V.Artisyuk, Y.Peryoga and K.Nikitin, Japan.

In France, Light Water Reactors (LWR) will continue to play a dominant role for generation of electricity during most of the current century. The first generation PWR 900 MWe were initially licensed to use low enriched uranium (LEU) oxide fuel. Subsequently, they were slightly adapted to accept upto 30% mixed uranium plutonium oxide (MOX) fuel loading in the core. A more efficient and economically acceptable plutonium management scheme is needed for PWRs till the fast reactors are commercialized. Accordingly, European Pressurised water Reactor (EPR) has been designed to allow loading of 100% MOX assemblies. The near term fuel programme aims at multiple plutonium recycling and high burn-up (≥60 GWd/t) keeping safety margins the same as for current UO_2-fuelled PWRs. The following three MOX based fuel concepts are under examination and are likely to be deployed commercially during 2015-25:

(i) APA and Duplex assemblies consisting of heterogeneous arrangement of PuO_2 in an inert matrix (eg. CeO_2) surrounded by UO_2 rods. A cermet fuel has also been envisaged where the PuO_2 particles are dispersed in zircaloy metal matrix.
(ii) CORAIL, using a heterogeneous arrangement of MOX rods (PuO_2 in a depleted UO_2 matrix) and UO_2 rods in a fuel assembly. The MOX rods could contain 0.25% U^{235} and as high as 11% Pu.
(iii) MOX-UE uses all homogeneous MOX rods with LEU in a standard PWR fuel assembly configuration

In France, a comprehensive overview has been made of the different fuel cycle options contemplated for High Temperature gas-cooled Reactors (HTR) application. HTRs are considerably adaptable to different fuel cycles, can accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design and have the inherent advantage of higher temperature of operation, improved efficiency and safety. Detailed studies have been made on the use of low enriched uranium cycle, MOX cycle, 'plutonium only cycle' (for consumption of excess weapons grade plutonium) and thorium-based cycles involving high enriched uranium (HEU), medium enriched uranium (MEU), low enriched uranium (LEU) and thorium plutonium cycles in HTRs. The low enriched uranium cycle clearly emerges as the most credible option for the first stage of HTR.

Atomic Energy of Canada Limited (AECL) has developed the innovative Advanced CANDU Reactor (ACR) of 700 MWe, an evolutionary adaptation of CANDU 6, but with a much lower capital cost and a lattice design that substantially reduces the coolant void reactivity. The ACR is a light-water cooled, heavy water moderated pressure-tube reactor with a smaller lattice pitch than the CANDU 6. It has a smaller calandria and requires less heavy water. Like CANDU reactors, the ACR has a small, simple fuel bundle design and on-power refueling scheme. Advanced fuel cycles, which have been examined for implementation in the ACR include: high burn-up LEU fuel cycles, MOX fuel cycles and thorium fuel cycles. The ACR shows exceptional promise in plutonium dispositioning as full-core MOX fuel cycles are possible. All ACR fuel cycle studies have taken advantage of the improved thermal hydraulic properties of the advanced 43-element CANFLEX fuel bundle. ACR is currently undergoing a pre-application licensing review in Canada and by NRC, USA.

In Russian Federation, new thermal reactors are planned to be put into operation upto 2020 at the rate of about 1 GW/year. The plutonium from the spent nuclear fuels from these reactors could be most efficiently utilized in commercial fast reactors from 2030 onwards. Accordingly, lead cooled innovative BREST fast reactor operating on closed plutonium fuel cycle has been developed, which would meet the INPRO requirements. Up to now, BREST 300 demonstration plant design is nearing completion and design activity of BREST 1200 commercial plant is underway. The essential features of BREST reactors are:

— Use of lead as coolant and reflector.
— Use of high density and high thermal conductivity (U,Pu)N as driver fuel.
— No uranium blanket and recycling of plutonium only as mixture with uranium.
— Co-location of reactor, reprocessing and fuel refabrication facilities.
— Generation of lower amounts of minor actinides and Rad waste

In Japan, feasibility study on commercialization of FBR and its fuel cycle by the year 2015 is underway. The FBR system would ensure sustainable energy and long-term use of nuclear power by recycling TRU fuels, burning minor actinides and transmuting long-lived fission products. Innovative technologies for the advanced fuel cycle system were reviewed and several candidate technologies were screened. Feasibility studies are underway for reprocessing and refabrication of MOX fuel for fast reactors in the form of 'pellet-pin' and 'vibro-packed pin' by aqueous recycle and non-aqueous recycle routes and metallic fuel by pyro electrochemical and casting route. An inherently Protected Plutonium Production (PPP) project is underway. This focuses the domain of uranium oxide fuel performances in commercial light water reactor environment by considering the following two straight forward options:

(a) Doping of minor actinides to 5% enriched uranium oxide.
(b) Increasing uranium enrichment in the initial fuel to about 20%.

Neutron capture of Np^{237} and decay of Cm^{242} lead to Pu^{238} accumulation, which has high neutron radiation and decay heat, thereby making the fuel inherently proliferation resistant.

The near term development of China's nuclear power will focus on matured technology, i.e. PWR and its fuel cycle and nuclear desalination. To cope with the challenges from deregulation of electricity market, the determining factor is to improve economic competitiveness of nuclear technology while maintaining its high level of safety. For mid- and long term, R&D programmes are underway for development of advanced PWR, fast reactor and HTR. An experimental Fast Breeder Reactor of 65 MW capacity is under construction and is expected to go critical in the year 2005. The HTR-10 MW, a PBMR attained criticality at the end of the year 2000 and was connected to grid in February 2003.

4. BACK END OF FUEL CYCLE

(i) "Global status and trends in spent nuclear fuel reprocessing" by J.S. Lee, IAEA
(ii) "Advanced reprocessing technologies" by B.Christiansen, D.Serrano, J.Serp, R.Malmbeck, J.P.Glatz, European Commission
(iii) "Role of pyro-chemical processing methods in defining eco-friendly advanced nuclear fuel cycles" by H.P. Nawada and K. Fukuda, IAEA
(iv) "Decommissioning of the Karlsruhe Reprocessing Plant (WAK)" by W.Pfeifer, J.Fleisch and G.Katzenmeier, Germany
(v) "Innovative approaches using DUPIC Processes" by K.S.Chun and M.S.Yang, Republic of Korea
(vi) "European ADS Programme – Present considered designs and perspectives" by Pierre D'hondt, Belgium

Management of spent fuel arising from nuclear power production is a crucial issue for sustainable development of nuclear energy. Till the year 2000, the cumulative discharge of spent fuel has been in the range of 230,000 MT heavy metals (hm) and the current annual discharge is in the range of 10,500-11000 MThm. So far, the cumulative reprocessing of spent fuel has been some 85,000 MThm of which more than 27,000 MThm is from LWRs, mostly at two commercial plants at La Hague, France and Sellafield, UK. Presently, the global annual reprocessing capacity of spent fuel is in the range of 5,000 MThm, mostly in France and UK and to a limited extent in Russia, Japan, India and China. In view of the large amount of spent fuel to be added to the cumulative inventory in the world, the significance of spent fuel management will continue to grow in the future. The total available reprocessing capacity by 2020 should increase with the introduction of new reprocessing plants planned in Japan, China and Russia. The technology utilized in spent fuel reprocessing facilities has improved rapidly to continuously adapt to the evolving fuel characteristics and other constraints, including stringent safety, security and regulatory requirements. Conventional reprocessing technology will continue to play an important role as an option for spent fuel management. In the interim period, till fast reactors are commercially deployed for global energy supply, plutonium would be recycled as MOX in LWRs, replacing a nearly equivalent amount of enriched uranium thereby minimizing uranium mining and enrichment operations. The conventional aqueous route, namely the PUREX process based on solvent extraction, is being modified and new processes like TRUEX and UREX 1 have been developed to meet the specific requirement of different countries. The pyro-chemical process, including electro refining of oxide, nitride and metallic fuel and molten salt extraction and fluoride volatilization are being relooked and critically examined in Japan and European Commission with emphasis on Partitioning and Transmutation (P&T) of minor actinides.

Taking advantage of Korean nuclear power strategy of having both PWR and CANDU for generation of electricity, the concept of Direct Use of spent PWR fuel In CANDU reactor (DUPIC) fuel cycle was initiated in 1991 jointly by KAERI, Republic of Korea, AECL, Canada and US DOS. The DUPIC fuel cycle has the inherent advantages of proliferation resistance, reduction of waste generation and saving of natural uranium resources. As part of DUPIC fuel cycle programme, feasibility study and experimental verification, including laboratory scale remote fabrication in hot cells and irradiation-testing in research reactor has been successfully completed and demonstration of DUPIC fuel performance in CANDU is planned during the period 2003-2007.

Decommissioning of the prototype Karlsruhe Reprocessing Plant, WAK, in Germany is underway. WAK was shutdown in 1991 after 20 years of operation during which time 210 metric tons of spent zircaloy clad UO_2 fuel from LWRs and HWRs had been reprocessed from which about 1.1 metric ton of plutonium was recovered and 70 m^3 High Active Waste Concentrate (HAWC) was generated. The aim of the project is to dismantle, decommission, decontaminate and recultivate the premises by 2010 as per the following programme:

— Dismantling of the main process building
— Parallel construction and operation of a vitrification plant to solidify HAWC
— After HAWC solidification, dismantling of HAWC storage facility and vitrification plants
— Conventional demolition of all buildings

The European Technical Working Group (ETWG) on Accelerator Driven System (ADS) has played a coordinating role at European level for Partitioning and Transmutation (P&T) and ADS development as a route for waste management and final disposal (FD) and closure of the back-end of fuel cycle. There is a need for first step demonstration of ADS at international level and coordinated R&D at European level with support from EC for implementing the following strategies recommended by ETWG:

— UO_2 in LWR + ADS + FD
— UO_2 in LWR + MOX in LWR + ADS + FD
— UO_2 in LWR + MOX in LWR + (MOX+MAs) in FBR + ADS + FD

Both critical and sub-critical reactors are potential candidates for dedicated transmutation systems. Sub-criticality is favorable and allows safe operation with a maximum load of minor actinides per unit.

5. INNOVATIVE NUCLEAR FUEL CYCLE CONCEPTS

(i) "Economic User Requirements" by Alan McDonald, IAEA
(ii) "Sustainability – Environment, Fuel Cycles and Waste" by E.Falck, IAEA
(iii) "Proliferation Resistance" by Tom Shea, IAEA
(iv) "On economic indicators for inter-comparison of nuclear and renewable energy sources" by N.Rabotnov, IAEA
(v) "INPRO User Requirements for Safety of Innovative Nuclear Reactors and Fuel Cycle Installations" by B. Kuczera, IAEA
(vi) "Methodology for Innovative Nuclear Technologies Assessment" by M. Khorochev, IAEA

Several experts in IAEA have critically reviewed INPRO user requirement from the point of view of economics, sustainability, environment, safety and proliferation resistance. The INPRO assessment methodology could be judiciously utilized by Member-States, policy makers and stakeholders to analyze the weakness and strengths of the different nuclear fuel cycle alternatives to improve and expand on-going national programmes. Sustainability of nuclear energy is the over arching principle that guides the development of basic principles and users requirements in the area of environment. Regarding sustainability from the environment point of view, two major issues arise: the acceptability of potential adverse environmental effects and the efficient use of renewable resources. For each one of these, a basic principle for environmental protection can be derived. These basic principles and their related user requirements are based on the view that an integrated and holistic assessment of the whole fuel cycle is required in order to assess and ensure its sustainability. Adverse environmental effects may arise from any components of the nuclear energy system and from any stage in its life cycle. Moreover, the design and operation of one component of the system can have a major influence on the environmental effects of other components. Therefore, the environmental performance of a proposed system must be evaluated as an integrated whole by considering the likely environmental effects of the entire collection of process, activities and facilities in the energy system at all stages of its life cycle. Besides, the fact that the adverse environmental effects of an innovative nuclear energy system must be within the current regulatory limits, there is an exception that its environmental performance will be better than that of existing systems. In order to achieve sustainability, a nuclear energy system must also make efficient use of any non-renewable sources involved such as fertile and fissile materials. User requirements have been factored in for the innovative nuclear energy system and the methodology used for its assessment has been developed. The basic assessment method will be a Material Flow Analysis (MFA) contributing to a Life-Cycle Impact Assessment (LCIA) of the innovative nuclear energy system.

Proliferation resistance is that characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices. The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures. Intrinsic proliferation resistance features are those features that result from the technical design of nuclear energy systems, including those that facilitate the implementation of extrinsic measures. Extrinsic proliferation resistance measures are those measures that result from States' decisions and undertakings related to nuclear energy systems.

INPRO defines seven basic principles of proliferation resistance as follows, where "intrinsic" features result from the technical design of nuclear energy systems and "extrinsic" measures result from States' decisions and undertakings related to nuclear energy systems.

— Proliferation resistance features and measures help ensure that future nuclear energy systems will continue to be an unattractive means to acquire materials for a nuclear weapons programme.
— Proliferation resistance could be enhanced when complementary and redundant features and measures provide defence in depth.
— Proliferation resistance will be most cost effective when an optimal combination of intrinsic features and extrinsic measures, compatible with other design considerations, can be included in a nuclear energy system.

— Proliferation resistance will be enhanced when taken into account as early as possible in the design and development of a nuclear energy system.
— Effective use of intrinsic proliferation resistance features facilitates efficient application of extrinsic measures.
— Extrinsic proliferation resistance measures, such as control and verification measures will remain essential, whatever the level of effectiveness of intrinsic features.
— From a proliferation resistance point of view, development and implementation of intrinsic features should be encouraged.

For intercomparison of economic indicators for nuclear and renewable energy sources like solar and wind, the 'capacity factors' of solar and wind plants have to be taken into consideration along with their 'installed or nominal capacity'. Solar installation and wind turbines have capacity factors in the range of 7-20% and 20-40% respectively which are much lower than the nuclear power plants, which has an average capacity factor in the range of 80-90%. Thus, for evaluating specific investment cost, average capacity factor should be used instead of 'nominal power capacity' in order to get a satisfactory economic indicator for intercomparison. Latest statistical research revealed that present PV costs are still about \$7 per peak Watt for on-grid systems and \$14 for off-grid systems, which means at least \$35, and \$70 per average Watt. The gap between solar and nuclear costs looks too wide to be ever filled. Their present booming expansion was made possible by heavy subsidies only, reducing their prices for private customers by almost an order of magnitude compared to actual costs. Sharp regular and random variations of renewable energy sources output indicate that they may only be operated as a supporting component together with much larger stable and reliable energy supplying systems. Hundred per cent of existing wind and solar power installations are less than ten years old and some eighty per cent – less than five years old. They are still to reveal the problems associated with aging and limited endurance in sharply varying and sometimes severe weather conditions affecting quality of transparent and reflecting optical elements and sealing tightness in solar installations and mechanical strength of hard working wind turbines machinery. For comparative analysis of nuclear energy with solar and wind, in terms of status, cost and performance, a closer, wider and critical evaluation is necessary.

6. CONCLUDING REMARKS

The diversity of nuclear reactor systems and fuel cycle options shows an extraordinary range. Hence, it is an open question as to whether all INPRO user requirements could be met with a small number of fuel cycle scenarios. It is felt that meeting the top-level economic requirement and funding R&D to meet the user requirements on a reasonable time scale could be very challenging. IAEA may consider future work in the following areas:

6.1. Continuation of studies on "Spent Uranium Fuel Management Options"

— Immediate reprocessing and plutonium recycling in LWRs and PHWRs.
— Immediate reprocessing and safe storage of plutonium for FBRs in future.
— Temporary storage and postponing reprocessing for future to recover plutonium for use in FBRs when they mature.
— Direct disposal or long-term storage followed by disposal.
— Development of ADS for spent fuel and waste management.

6.2. Assessment of present day technologies in uranium fuel cycles and its intercomparison with advanced dry, aqueous and pryo recycling technologies.

6.3. Assessment of partitioning and transmutation (P&T) issues with respect to:

- Benefits on long-term radiological impacts of the final disposal of the ultimate wastes and its environmental effects.
- Additional dose to workers
- Proliferation resistance
- Additional Cost

6.4. Assessment of thorium fuel cycles with respect to:

- Resource Utilization ('Th' more abundant in nature than 'U')
- Inherent Proliferation Resistance (presence of U^{232} in U^{233})
- Out-of-Pile Properties & In-Pile Behaviour ('Th'-based fuels have better physical and thermophysical properties compared to its 'U'-based counterparts) Reduction of High-level Waste GenerationPartitioning & Transmutation of 'Th' Cycles

6.5. Sensitivity Analysis of different fuel cycle options in changing scenarios with respect to:

- Resource Availability,
- Technological Advances and
- Other Emerging non-Carbon energy systems (eg. wind & solar)

6.6. Preparing Manuals, offering technical support & initiating public information programmes concerning siting, construction, operation & closure of high-level waste repositories in deep clay, granite or salt formations

6.7. Making INPRO a standard IAEA project, not an extra budgetory one, for the benefit of all Member States

7. ACKNOWLEDGEMENTS

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OTHER TOPICS

(Session 6.3)

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THE ROAD MAP FOR A FUTURE INDIAN NUCLEAR ENERGY SYSTEM

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Abstract. The three stage Indian nuclear programme aims to establish a sustainable nuclear energy system mainly based on closed nuclear fuel cycle and thorium utilization. The technology for the first stage, based on Pressurized Heavy Water Reactors (PHWRs), has already reached a level of maturity, and a commercial fast breeder reactor programme, comprising the second stage, is due to start soon with the construction of Prototype Fast Breeder Reactor (PFBR). A road map for the entire integrated future Indian nuclear energy system, has been recently drafted with the following major long term objectives: (1) Providing long-term energy security by using available nuclear fuel resources in an optimal manner; (2) Enhancing safety of nuclear system to enable its siting even in close vicinity of populated regions; (3) Minimising the long-lived radioactive wastes, eventually to reach a situation of `radiation balance'; and (4) Apart from the main thrust on nuclear power generation, meeting the needs of non-grid based electricity supply, generation of fluid fuels for transportation applications, compact power packs with long life cores and desalination. The main new components of this integrated nuclear energy system are envisaged now to comprise the following: (1) Advanced Heavy Water Reactor (and other thorium fuelled reactor systems); (2) Compact High Temperature Reactor, (3) Accelerator Driven System, (4) Advanced fuel cycle (front end and back end) facilities. The paper briefly provides a technical description of the above-mentioned four components and presents the results of a study to indicate the potential of the road map to meet its stipulated major long-term objectives.

1. INTRODUCTION

Globally, the abundance of thorium is thrice that of uranium. It is generally agreed that closed fuel cycle and thorium utilisation will be needed for long term sustainability of energy supply in the world.

In particular, the known Indian uranium reserves cannot support even a modestly sized nuclear power programme, if this fuel is used in an open cycle. On the other hand, the Indian reserves of thorium are one of the largest in the world. Considering the domestic availability of nuclear fuel resources India, a large country with a rapidly growing energy demand, has a high priority for development and deployment of closed nuclear fuel cycle and thorium utilisation based nuclear energy systems.

With the expected growth in the nuclear power programme, future nuclear energy systems are required to meet progressively increasing needs for further enhanced safety features and sustainability from environmental considerations, while remaining economically competitive.

The Indian nuclear energy programme follows a well laid out road map to achieve the afore-mentioned objectives.

2. THREE STAGE NUCLEAR PROGRAMME

To ensure long term availability of nuclear energy in a sustainable manner, taking cognisance of its resource position, India has followed the closed fuel cycle and chalked out a three-stage nuclear power programme (Fig. 1) based on uranium and thorium [1].

The three stages of this programme comprise:

(i) Natural uranium fuelled Pressurized Heavy Water Reactors
(ii) Fast breeder reactors utilising plutonium based fuel
(iii) Advanced nuclear power systems for utilization of thorium

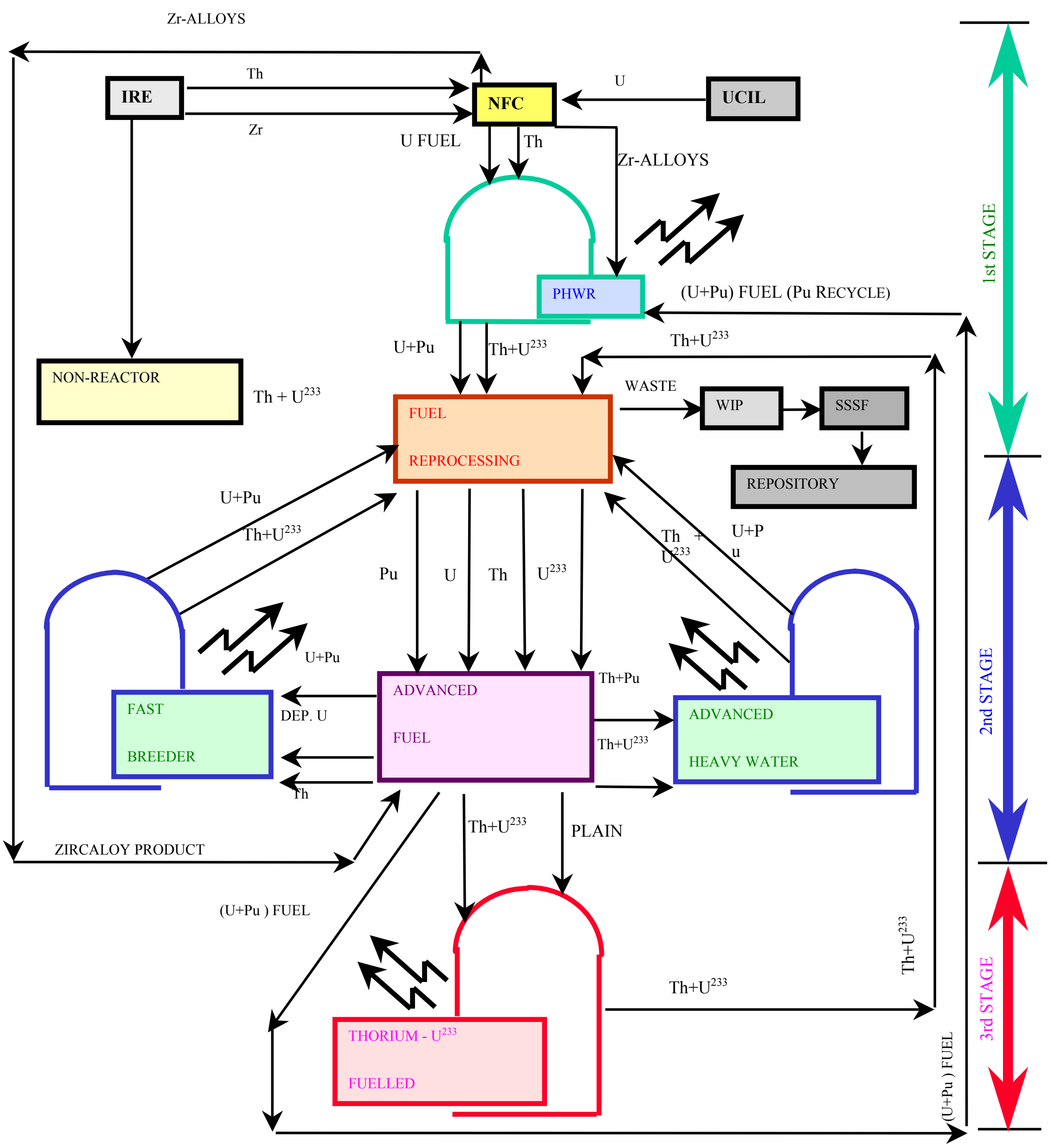

FIG. 1. Three stages of the Indian nuclear power programme.

Here it must be mentioned that the three stages have considerable, and necessary, overlaps in the time frames of their development and deployment. It should be noted that though the three stages comprise the main stream of the Indian nuclear power programme for electricity generation, the programme has enough flexibility to accommodate some variants and additional elements, including light water reactors, for augmenting the nuclear power base as needed. In addition, flexibility also exists for the use of nuclear energy for applications other than generation of grid-based electricity.

3. CURRENT PROGRAMMES

3.1. Natural uranium fuelled pressurized heavy water reactors

Pressurized Heavy Water Reactor (PHWR) is an excellent reactor system for using natural uranium as fuel. Thus the first stage of Indian programme consists of setting up of a series of PHWRs. Presently the PHWR programme is in industrial domain with twelve PHWRs among fourteen operating reactors, a fuel fabrication plant and three reprocessing plants. Two 540 MWe PHWRs and four 220 MWe PHWRs are under construction, and few more units have been planned. All aspects of PHWR technology have been mastered. With the standardisation of the units and operating experience gained, the performance of the plants has improved tremendously with overall capacity factors steadily rising from 60% in 1995-1996 to 90% in 2002-2003.

PHWR technology has already reached maturity. Further improvements will concentrate on reducing plant gestation period, reducing capital cost and O&M cost, further developing in-service inspection and repair technologies, up-gradation of control and instrumentation systems, ageing management and plant life extension, and incorporation of improved fuel designs including (U-Pu) Mixed Oxide (MOX) and high burn-up fuels [2]. For the new 540 MWe plants to be constructed, the Nuclear Power Corporation of India Limited (NPCIL) is working towards introducing boiling in few of the high power channels in the core. The boiling, to the extent of three percent, offers a possibility of up-rating the capacity from 540 MWe to 700 MWe.

3.2. Fast breeder reactors

The second stage of the Indian nuclear programme aims at setting up of Fast Breeder Reactors (FBR) for power production and fissile material multiplication. The second stage started with setting up of a Fast Breeder Test Reactor (FBTR) at the Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam to understand and master the fast reactor technology. FBTR has been running successfully for the past 15 years, utilizing a unique and indigenously developed uranium- plutonium carbide fuel. The fuel has recently crossed peak burn-up of 100 MWd/kg [3]. The experience gained in fast reactor technology has manifested in the design and development of a 500 MWe Prototype Fast Breeder Reactor (PFBR). Plutonium and depleted uranium oxide will be utilized in PFBR and the fuel fabrication for the initial core will be done at Advanced Fuel Fabrication Facility (AFFF), Tarapur.

Further developments will focus on reducing the capital cost, O&M costs and fuelling cost while improving safety [2]. These involve R&D in the following areas:

— Increased design life of plant
— High steam temperatures increasing thermal efficiency
— Better materials and structural analysis capability

— Development of high burn-up fuel (100 MWd/kg to 300 MWd/kg) to reduce fuel costs, fuel handling costs and rad waste.
— Fast reactor fuel reprocessing and fuel fabrication technology development
— Fuel cycle for short doubling time.

3.3. Light water reactors

To jump-start the nuclear power programme, two Boiling Water Reactors were set up at Tarapur in the late sixties, and these are still in operation. With a view to accelerate the nuclear power generation capacity, few light water reactors are planned to be set up in addition to the afore-mentioned self-reliant three stage programme. Two 1000 MWe units are under construction at Kudankulam, Tamil Nadu with Russian collaboration. Department of Atomic Energy (DAE) has planned a programme for electricity generation of 20,000 MWe by the year 2020, which includes HWRs, FBRs and LWRs.

3.4. Thorium in existing reactor systems

Thorium cycles are feasible in all the existing thermal reactors and fast reactors as well. In the short term, it should be possible to incorporate the thorium fuel cycle in existing reactors without necessitating major modifications in the engineered systems, reactor control and the reactivity devices.

Studies have been carried out for incorporation of three different thorium fuel cycles in PHWRs. These are the Self Sustaining Equilibrium Thorium cycle (SSET), the high burn-up open cycles and the Once Through Thorium Cycle (OTT) [4].

At present thoria bundles are used in Indian PHWRs for achieving the initial flux flattening in the core. This represents a unique way of utilizing thorium without any loss of burn-up in UO_2 fuel. The ThO_2 bundles have been irradiated in Kakrapar-1 & 2 (70 bundles), Rajasthan-2, 3 & 4 (88 bundles), Kalpakkam (4 bundles) and Kaiga-1 & 2 (70 bundles). The samples from one of these bundles have been analyzed for fission products and uranium isotopic composition. These bundles will be used for the development and demonstration of reprocessing of ThO_2 bundles irradiated in power reactors, and will provide U^{233} for the development of fabrication technology and irradiation experiments.

4. ADVANCED NUCLEAR POWER SYSTEMS FOR UTILIZATION OF THORIUM

The third stage of the Indian nuclear power programme envisages setting up of reactors based on Thorium-U^{233} cycle. A road map for the entire integrated future Indian nuclear energy system, has been drafted with the following major long term objectives:

a) Providing long term energy security by using available nuclear fuel resources in an optimal manner.
b) Enhancing safety of nuclear system to eliminate any necessity for counter measures in public domain, following any accident in the plant. This implies possibilities for siting reactors in close vicinity of populated regions.
c) Minimizing the long-lived radioactive wastes, eventually to reach a situation of `radiation balance'.

d) Apart from the main thrust on nuclear power generation, meeting the needs of non-grid based electricity supply, generation of fluid fuels for transportation applications, compact power packs with long life cores, and desalination.

The main new components of this integrated nuclear energy system are envisaged now to comprise the following [5]:

— Advanced Heavy Water Reactor (and other thorium fuelled reactor systems)
— Compact High Temperature Reactor
— Accelerator Driven System
— Advanced fuel cycle (front end and back end) facilities

4.1. Advanced heavy water reactor (AHWR) [6]

The main objective for development of AHWR is to demonstrate thorium fuel cycle technologies, along with several other advanced technologies required for next generation reactors, so that these are readily available in time for launching the third stage.

AHWR is a 300 MWe, vertical, pressure tube type reactor cooled by boiling light water and moderated by heavy water. The reactor is fuelled with (U^{233}-Th) MOX together with (Pu-Th) MOX. AHWR is nearly self-sustaining in U^{233}. The design of AHWR is fine tuned towards deriving most of its power from thorium based fuel, while achieving negative void coefficient of reactivity.

Heat removal from core is achieved by natural circulation of coolant. As shown in Fig. 2., water-steam mixture from core rises through the tail pipes to enter the steam drum. In steam drum, separated water at saturation temperature mixes with feed water and flows down the downcomers to the inlet header. From inlet header, water enters core through inlet feeder pipes.

AHWR incorporates several advanced features to increase its safety, reliability and economics. These are enumerated below:

— Natural circulation heat removal under normal operation and shutdown conditions
— Low core power density
— Slightly negative void coefficient of reactivity
— Direct spray of Emergency Core Cooling System (ECCS) water into fuel pins during Loss Of Coolant Accident (LOCA)
— Advanced accumulator with fluidic device for ECCS
— Gravity driven cooling system ensuring core cooling for three days following LOCA, without operator intervention
— Passive containment cooling and isolation
— Utilization of moderator heat
— Utilization of low grade heat for desalination

The AHWR fuel comprises a 54 pin composite fuel cluster containing 24 $(Th\text{-}Pu)O_2$ pins and 30 $(Th\text{-}U^{233})O_2$ pins. It is planned to produce the requisite amount of U^{233} by in-situ generation in the reactor. Hence it is planned to start with an initial core containing fuel with all 54 pins containing $(Th\text{-}Pu)O_2$. The U^{233} in the spent fuel fuel will be recycled back to the reactor after reprocessing.

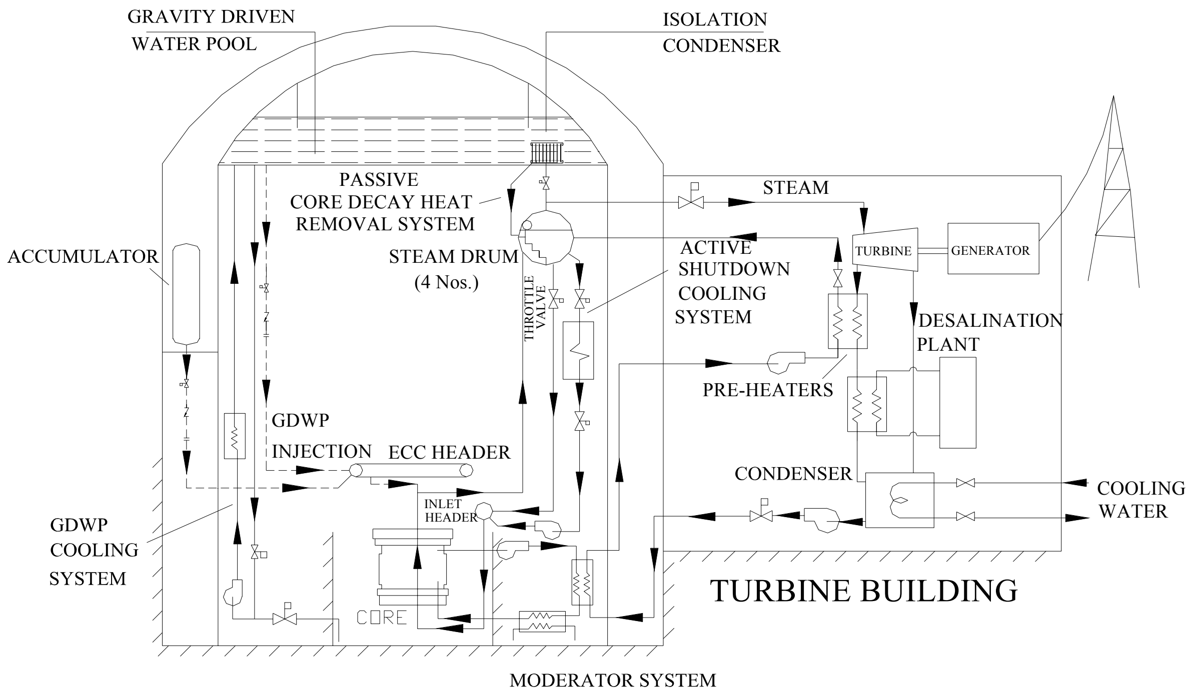

FIG. 2. Schematic of advanced heavy water reactor.

In order to enhance the safety of nuclear systems, a three pronged approach is followed. In the first step, the systems are so designed that probability of occurrence of an accident is reduced significantly below the level existing in current reactors. Next, even for very low probability events, the consequences are mitigated or reduced significantly so that no accident management is foreseen. Finally, the leakage through multiple barriers to the environment is also significantly reduced. The AHWR relies heavily on passive processes and components for its operation and accident mitigation.

A number of facilities have been either already built or are under construction or planning to validate the evolutionary concepts used in AHWR. The target core damage frequency is 10^{-7}/year or less. For achieving this, a series of potential high consequence events were visualized even when these events are Beyond Design Basis Accidents (BDBA) according to current standards. The safety features are so engineered that the consequences are mitigated. One such class of events is Anticipated Transients Without Scram (ATWS). Such events are analyzed in detail, even though, according to current standards, with two independent and diverse safe shut down systems, these events are BDBA. Results of safety analyses show that for a host of such events the fuel temperatures remain below their failure limits.

For ensuring very limited leakage to environment during accidents, a double containment is used. All potential leakage points are strengthened.

These added safety features aim to qualify the design of AHWR for its siting close to populated regions without any need for evacuation planning.

4.2. Compact high temperature reactor [7]

Nuclear reactors will need to be increasingly utilized in the future for non-electrical high temperature process heat applications including production of hydrogen or secondary hydrocarbons as a substitute for primary fossil fuel, and for serving as components of compact power packs in remote areas not connected to grid system. To meet such needs a Compact High Temperature Reactor (CHTR) (Fig. 3.) is being developed in Bhabha Atomic Research Centre (BARC).

CHTR is being designed with the following design guidelines

— Use of thorium based fuels
— Passive core heat removal by natural circulation of liquid heavy metal coolant
— Passive power regulation and shutdown mechanism.
— Passive rejection of entire heat to the atmosphere under accidental condition
— Compact design to minimise weight of the reactor

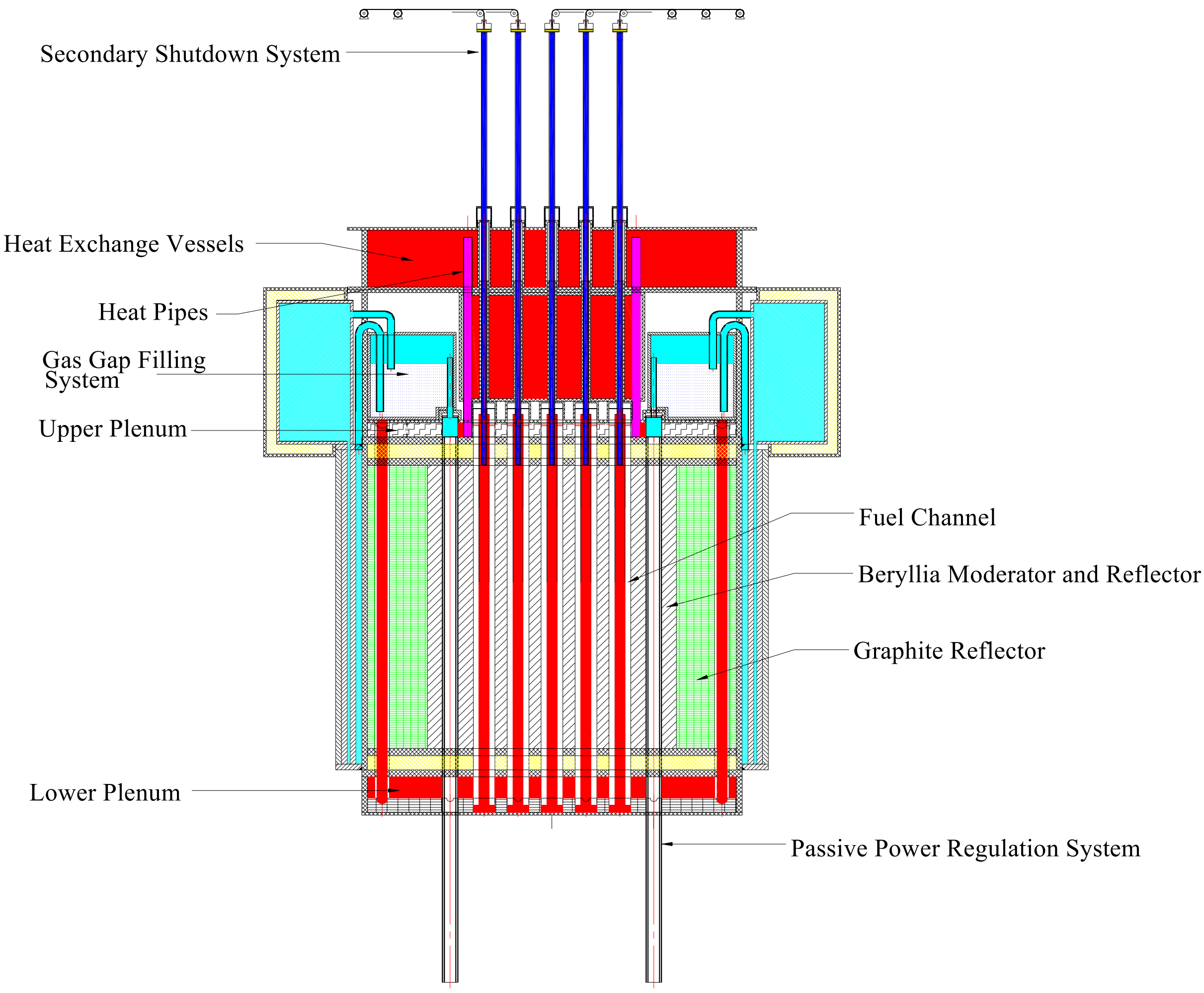

FIG. 3. Schematic of compact high temperature.

Based on these guidelines a conceptual design of CHTR has been worked out. In the current stage of design, the reactor core consists of nineteen prismatic beryllium oxide (BeO) moderator blocks (distributed in 1-6-12 arrangement). These blocks contain centrally located carbon fuel tubes. Each fuel tube carries within it the fuel inside 12 equi-spaced longitudinal bores. The fuel is in the form of pellets containing coated fuel particles. The central bore of the fuel tube serves as coolant channel. Eighteen blocks of beryllium oxide reflector then surround the moderator blocks. The beryllium oxide reflector blocks are surrounded by graphite blocks as additional reflector.

To remove heat from the reactor core, a lead-based liquid metal coolant flows by natural circulation between the top and bottom plenum, upward through the fuel tubes and returning through downcomer tubes. From the upper plenum of the reactor, heat pipes transfer heat to heat transfer interfaces of heat utilizing systems. These heat utilizing system interfaces provide the required environment, heat transfer area and interface hardware for energy conversion systems. The reactor has been provided with passive reactor power regulation system. The system uses core outlet coolant temperature as a driver to passively cause an extent of absorber insertion.

CHTR has been provided with systems to reject entire core heat to the atmosphere by passive means during postulated accident conditions. These heat removal systems operate to prevent the temperature of the core and coolant from increasing beyond a set point.

Some of the technologies required for such a high temperature reactor will also be useful for possible implementation in a future version of AHWR, to further improve the economic performance of the latter. Molten lead related and high temperature application specific technologies are important for the accelerator driven systems.

4.3. Accelerator driven subcritical systems (ADSS) [8, 9]

Accelerator driven subcritical systems (ADSS) throw open several attractive possibilities for extending the Indian nuclear power programme. ADSS can, in principle, be used to produce several times more electrical energy than that is required to run the accelerator. Such systems have a potential to convert fertile materials to fissionable materials, and to transmute the highly radioactive waste from conventional nuclear power plants to shorter lived radio-nuclides, which do not require a very long term storage under surveillance.

The challenges involved in the design of very large accelerators, and coupling them to sub-critical cores, are quite substantial and are matters of intense R&D effort in several countries. ADS aim to contribute to improvement in the overall energy scenario in the long-term time frame.

In India, a beginning has been made in acquiring the necessary expertise to design and build linear accelerators as well as cyclotrons in India. As a part of the roadmap for the third stage of the Indian nuclear power programme, a set of milestones has been identified, along the way of the development the technologies relevant for an accelerator driven system. These milestones ar enumerated below:

— To develop high-energy neutron source, which can be used in the Critical Facility under construction at BARC, for physics experiments. This will help in quickly validating relevant neutronic data, which can be used for the design of accelerator driven systems.

— To develop a spallation neutron source, utilizing molten heavy metal as target, along with a moderate sized accelerator, and use it in a research reactor core for additional R&D on the coupling of this source with a sub-critical core.
— To design and develop an accelerator driven fertile to fissionable material converter, where the basic objective is to produce fissionable material from Th^{232}/ U^{238} without generating electricity.
— To build full-fledged accelerator driven systems, for electricity generation, fissionable material production, and nuclear waste incineration applications. Such systems can then work synergistically with the remaining components of our nuclear power programme.

There are several key technological areas in which R&D is needed to reach the afore-mentioned milestones. A detailed plan is being compiled to identify and define the scope of each of these R&D activities. Some elements of this plan are listed below:

— Development of accelerators in the GeV range and current of the order of milli-amperes
— Development of special materials for spallation source and other structural features
— Accelerator driven spallation source design studies
— Nuclear data relevant for materials and energy spectrum of ADS.
— 14 MeV neutron sources for experimental studies in a sub-critical facility.
— Subcritical Reactor physics with spallation neutrons
— Development of fertile material conversion blankets

Normally, accelerators with sufficiently high current would be required for ADSS. Such systems are currently under development. A one-way coupled fast-thermal system has been proposed to get around this problem at least partially. An interesting aspect of this design of ADSS is that due to high decoupling of booster and reactor, the specific power density in the main thermal reactor is very low. Specific power density in AHWR is also low on account of natural circulation. Therefore, a modified AHWR core can be adapted for the thermal portion of such ADSS.

4.4. Advanced fuel cycle facilities

In order to support the fuel cycle of AHWR, several technologies have to be brought to a level of maturity comparable to that for uranium fuel cycle, and some new technologies have to be developed. AHWR is nearly self-sustaining in U^{233}, which makes the fuel cycle for AHWR a closed one, making reprocessing and refabrication necessary for sustained operation. U^{232} is an impurity associated with U^{233}. Its decay products are hard gamma emitters, which pose radiological problems during fuel fabrication. Hence, the U^{233} based fuel needs to be fabricated in shielded facilities. For this purpose the automation and remotization technologies used in the current fuel fabrication (UO_2 and MOX) plants need to be considerably enhanced. In addition, work is being carried out in BARC in developing alternative fuel fabrication technologies amenable for remotization like sol-gel micro sphere pelletization, vibro-packing, pellet impregnation and advanced agglomeration techniques. The above technologies aim at tackling the radiation problem by remote fabrication and modifications in the process. Developmental work on various fabrication processes is carried out at AFFF, Tarapur [10]. Fig. 4 shows some of the products fabricated at AFFF – annular pellet and granules produced by advanced agglomeration technique.

The development of laser isotopic separation (clean up of U^{232} from U^{233}) process is also being done.

Annular Pellet

Granules by Advanced Agglomeration Technique

Impregnated Thoria-Urania Microspheres after Calcination and Reduction

FIG. 4. Developmental work at advanced fuel fabrication facility (AFFF).

The back end of thorium cycle also presents technological challenges. Thorium, being an inert and stable material, poses difficulty in dissolution during spent fuel reprocessing. Laboratory scale parametric studies have been carried out for the dissolution of thoria with different additives like solvents (HF and NaF), dopants (Magnesium oxide) and aqueous vs. microwave heated dissolution. The Thorex process is used for reprocessing of thorium dioxide, which is similar to the Purex process followed for reprocessing of uranium. Facility for Uranium-233 Separation (FUS), a plant for separation of uranium from thorium is coming up at BARC, Trombay. Fig. 5. shows operation in glove boxes in FUS.

The reprocessing of AHWR spent fuel will need to be done in three streams containing uranium, thorium and plutonium. The reprocessing processes till now have been concerned essentially with the extraction of fissile materials. In the fuel cycle for AHWR, the extraction and utilization of thorium for recycle is also being explored, in which case the Thorium-228 (Th^{228}) associated with Thorium-232 (Th^{232}) poses radiological problems during refabrication. Extraction of thorium, uranium and plutonium has been carried out in a pilot plant scale using fuel bundles irradiated in research reactors.

In India, the experience on reprocessing of thoria fuel is restricted mainly to aluminium-clad thorium and ThO_2 fuel irradiated in CIRUS reactor. Fuel rods irradiated up to a level of 1.2 kg of U^{233}/te of thorium and cooled for more than 2 years were reprocessed in a pilot scale test facility at BARC and IGCAR. A plant for zircaloy clad thoria fuel is being constructed at BARC. Till now aqueous reprocessing has been carried out using Tributyl phosphate (TBP) as the extractant. The scope of future work lies in the development of alternate solvents, pyro reprocessing technology and reprocessing of fuel with multiple recycling.

The recovery of long-lived actinides and fission products of value could make nuclear power eco-friendly and improve economics in the long run. Partitioning and transmutation involves processing the waste to extract long-lived radio nuclides, which may then be irradiated in a fast reactor or an ADSS to yield products with shorter half-lives, thereby reducing the time required for their isolation from the environment. The ultimate aim is to achieve high degree of separation of plutonium and actinides so that the high level waste does not carry the burden of actinides. However, the issue of increased volume of secondary waste needs to be addressed while considering the partitioning of actinides.

FIG. 5. Facility for Uranium-233 separation (FUS).

5. SOME RESULTS OF PHYSICS STUDIES PERTAINING TO AHWR FUEL CYCLE

5.1. Production of U^{232}

As already mentioned, thoria bundles are being used for initial flux flattening in Indian PHWRs. One bundle with a discharge burnup of around 11500 MWd/teHM has undergone Post Irradiation Examination (PIE). A sample of the pellet was dissolved and analyzed for uranium isotopic composition. It contained nearly 500 ppm of U^{232}. ENDF-B VI cross-sections were condensed to effective one group cross-section and simulations were carried out with ORIGEN-2 code, which is a point depletion and burn-up code and which keeps track of generation and decay of more than one thousand nuclides present in the reactor core [11, 12]. With this approach a good match was obtained with the PIE results with respect to U^{232} composition.

5.2. Production of minor actinides

The computer code ORIGEN-2 was also used to evaluate the generation of minor actinides. In order to compare the results, the calculations have been performed for PHWR and AHWR fuel. For AHWR, the one–group effective cross-sections used in ORIGEN-2 are based on the same spectrum as PHWR. These results are summarized in Table I.

The AHWR fuel cluster consists of two types of fuel pins. The calculations for the two types of fuel pins have been performed separately at relevant burn-ups and then these have been combined. Table I gives the production of minor actinides, their radioactivity, toxicity index and toxicity potential in air as well as water for PHWR and AHWR fuel after a cooling period of 10 years following discharge.

Table I. Fuel cycle studies – for fuel with 10 yrs cooling

		PHWR	AHWR		
Fuel		Natural Uranium	24 pins $(Th-Pu)O_2$	30 pins $(Th-U^{233})\ O_2$	Full cluster
Specific Power, kW/kg		25.81			14.62
Average Burnup, MWd/te		6500	27243	21406	24000
Concentration of minor actinides, g/TWhe		2.75×10^{3}	6.07×10^{3}	1.13×10^{1}	3.07×10^{3}
Radioacivity, Ci/te/TWhe		7.34×10^{3}	2.15×10^{4}	5.31×10^{-1}	1.09×10^{4}
Toxicity Potential	Air, Sv/te/TWhe	1.03×10^{10}	3.10×10^{10}	1.97×10^{6}	1.56×10^{10}
	Water, Sv/te/TWhe	5.32×10^{7}	1.52×10^{8}	1.38×10^{4}	7.67×10^{7}
Toxicity Index	Air, m^3/te/TWhe	3.62×10^{16}	1.04×10^{17}	1.33×10^{13}	5.25×10^{16}
	Water, m^3/te/TWhe	2.16×10^{10}	5.16×10^{9}	5.90×10^{5}	2.60×10^{9}

Toxicity index gives the amount of water/air required to dilute the toxicity of nuclide to the level of MPC (Maximum Permissible Concentration) and the toxicity potential is the radioactivity in Curies multiplied by the DCI (Dose Coefficient of Intake). In order to compare these results for different systems, the results have been presented per unit energy (per TWh electrical) produced.

The production of minor actinides per TWhe in AHWR is slightly higher than in PHWR fuel, despite using thorium-based fuel. This is mainly due to use of plutonium enrichment in 24 pins of the fuel cluster which gives rise to higher production of americium and curium isotopes. The results for the $Th-U^{233}$ pins reinforce the known fact that with $Th-U^{233}$ fuel cycle, production of minor actinides and associated radioactivity is lower by several orders of magnitude, as compared to uranium fuel. As already indicated, the AHWR lattice lends itself to serve as a subcritical thermal core for the ADSS, and in such a configuration the need for having plutonium bearing fuel is totally eliminated.

5.3. Reduction in fissile plutonium content

India proposes to utilize plutonium to increase its energy potential. The AHWR uses thoria as the base matrix for the fissile isotopes to convert the fertile thorium to fissile U^{233}. The physics studies carried out have shown that the initial content of 75% fissile plutonium in the fuel cluster reduces to 28% at discharge. This is in agreement with several international studies which have shown that thoria is one of the best options as a carrier for burning plutonium.

6. SUMMARY

Harnessing nuclear energy from thorium based reactor systems is envisaged in the third stage of Indian nuclear power programme. This will be achieved with an integrated nuclear energy system comprising of AHWR, CHTR, ADSS and fuel cycle technologies required for thorium utilization.

7. REFERENCES

[1] DEPARTMENT OF ATOMIC ENERGY MONOGRAPH, Nuclear Power for National Development: An Indian Perspective, (2001).

[2] DEPARTMENT OF ATOMIC ENERGY, INDIA, Long Term Vision of the Department of Atomic Energy, Document No. 7, (November 2001).

[3] IGCAR News in www. igcar.ernet.in.

[4] SINHA, R.K. AND ANANTHARAMAN K., "Thorium as the Future Source of Power for India", INSAC-1999, (Annual Conference of Indian Nuclear Society), Mumbai (1999).

[5] DEPARTMENT OF ATOMIC ENERGY, INDIA, Shaping the Third Stage of Indian Nuclear Power Programme, Document No. 3 (July 2001).

[6] SINHA, R.K., et al, "Design and Development of AHWR - The Indian Thorium Fuelled Innovative Nuclear Reactor", INSAC-2000, (Annual Conference of Indian Nuclear Society), Mumbai (2000).

[7] KELKAR, P.P., et al, "Conceptual Design Report on Compact High Temperature Reactor", Report no. CHTR/RED/2002/13, (March 2002).

[8] KAPOOR, S. S., "Accelerator-Driven Subcritical Reactor System (ADS) for Nuclear Energy Generation", HIAT-2001, (Ninth Int. Conf. on Heavy Ion Accelerator Technology), New Delhi, (2002).

[9] KAKODKAR, A., "Shaping the 3rd stage of Indian Nuclear Programme", INSAC-2001, Annual Conference of the Indian Nuclear Society, Presented at the Annual Conference of Indian Nuclear Society, , Indore, (2001).

[10] ANANTHARAMAN, K., RAMANUJAM, A, KAMATH, H.S., MAJUMDAR, S. AND VAIDYA, V.N.,"Thorium based fuel reprocessing & refabrication technologies and strategies", INSAC-2000, (Annual Conference of Indian Nuclear Society), Mumbai, (2000).

[11] CROFF, A. G., "ORIGEN2.1, Isotope Generation and Depletion Code, Matrix Exponential Method", Report ORNL-5621 (1980).

[12] CROFF, A. G., "ORIGEN2: A versatile computer code for calculating the nuclide composition and characteristics of nuclear materials", Nuclear Technology, **62**, (1983), 335-352

SVBR-75/100 — LEAD-BISMUTH COOLED SMALL POWER MODULARFAST REACTOR FOR MULTI-PURPOSE USAGE

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Abstract. Today's nuclear power is in the state of an intrinsic conflict between economic and safety requirements. This fact makes difficult its steady development. One of the ways of finding the solution to the problem is use of modular fast reactors cooled by lead-bismuth coolant that has been mastered in conditions of operating reactors of Russian nuclear submarines. Based on this experience, the small power fast reactor for multi-purpose usage (SVBR-75/100) has been developed. The small power reactors make it possible to fabricate the whole reactor at the factory and deliver it to the NPP site in practical readiness by using any kind of transport including the railway. Reactor installation SVBR-75/100 was designed in compliance with a conservative approach. This approach presumes to use to the maximal extent the technical solutions and temperature parameters, which have already been verified by operating experience. Further, when a one-through steam-generator producing over-heated steam is used, technical and economical parameters will be considerably improved. Technological maintenance of fissile materials non-proliferation has been ensured at all lifetime stages. Refuelling in developing countries has not been provided. On ending the lifetime (~10 years), the reactor will be transported to the Supplier-country along with the core in the “frozen” coolant.

1. INTRODUCTION

To achieve the mature phase of nuclear power (NP) development with replacement of approximately 50 % of fossil fuel sources for electricity production, it is projected that several nuclear power technologies (NPT) including the nuclear power plants (NPP) of a certain type and a corresponding nuclear fuel cycle (NFC) will be superseded.

The NPT that best meets the requirements of a current stage of NP development in a certain country will dominate at each stage of NP development.

Probably, the duration of each stage necessary to achieve the mature phase of the NP would take many decades, which is caused by significant sluggishness of development of any new NPT.

The mature phase of the NP will be featured by:

— Domination of fast reactors (FR) operating in the entirely closed NFC;
— The most complete realization of the inherent safety principles;

— Finding the practical solution to the problem of handling long-lived radioactive waste (RAW);
— Maximum of technological maintenance of nuclear fissile materials (NFM) non-proliferation.

As the alternative energy sources exist together with the NPPs, at each stage of NP development in conditions of a liberalized electricity market, the NPPs must be competitive with the heat power plants (HPP) using fossil fuel.

The NPT will be superseded by the other one in case:

— The new NPT showing the better technical parameters appears;
— The NPP competitiveness is lost due to the higher paces of increase of the nuclear fuel costs at exhaustion of cheap nature uranium sources comparing with the paces of increase of the fossil fuel costs (that is low probable) or due to the higher costs of the NPPs and NFC caused by imposing the more stringent requirements.

For example, if the regulation authorities specify the requirements for non-proliferation technological maintenance as well as high safety requirements, the NPT competitiveness would be noticeably affected.

At the same time, increase of the cost of any source used by a certain NPT will stimulate both the exploration and development of this source (e.g. uranium) and launching works on changing over to the new NPT using this source more effectively (e.g. NFC closing) or using the new source (e.g. thorium).

However, at the electricity market the NPP competitiveness with HPPs using fossil fuel is not sufficient enough to ensure self-financing of NP development with a pace providing increase of the NP share in the total production of electricity generated by all kinds of power sources.

This is caused by the fact that at the liberalized electricity market in the developed countries the rate of electricity costs is decreasing due to excess of generating power capacities. In Russia the investment potentials of the NP have been limited by regulated cost rates, increase of an annual electricity production cost, and a value of total annual electricity production and they cannot cover the long-term investment needs consisting of the costs for:

— Increasing the loading factor (LF) of the NPP;
— Enhancing safety of the first generation units;
— Extending the life time of the units over the designed ones;
— Constructing the units of a high and medium extent of readiness;
— Decommissioning the NPP units with an expired lifetime;
— Constructing the new replacing power capacities compensating for the decommissioned units;
— Constructing the new NPPs providing a desired pace of NP development.

Along with it, in conditions of a market economy the NP cannot rely on any noticeable support of the state financing.

This fact put forward the requirement for the NPP competitiveness at the market investments (including those based on the credit repayment).

Realization of this requirement needs for considerable reduction of specific capital costs of NPP constructing and construction terms making these parameters close to those of the modern steam-gas HPPs. Besides, to reduce the investor's risk, it is necessary to considerably improve the safety level in order to eliminate the severe accidents like Chernobyl one. This is also necessary for ensuring an acceptance of the NP by public opinion when its scale is considerably increased.

It is very difficult to solve this problem on the basis of evolutionary improvement of the traditional NPP designs with thermal neutron reactors because an intrinsic conflict between economic and safety requirements is peculiar to such reactor installations (RI). This cause a necessity of constant increasing the unit power of the reactors that results in increasing the total investments, increasing the construction terms and reducing an investment attractiveness of the design.

Besides, the existing thermal neutron reactors cannot provide long (hundreds of years) development and functioning of the NP due to the low efficiency of using natural uranium power potential even in the closed NFC that makes electricity production more expensive at low prices of natural uranium.

The solution to this problem can be find by using an innovative NPT that uses a new type of the fast reactors (FR) which must not build up plutonium with a short doubling time (this task has lost its actuality).

This enables chemically inert heavy liquid-metal coolants (HLMC) (instead of sodium) with high boiling point to be used for heat removal, i.e.: eutectic lead-bismuth alloy (45 % of Pb, 55 % of Bi) that has been mastered in conditions of operating the nuclear submarines' (NS) reactors of Russian Navy [1] and currently mastering lead coolant.

Available reference information on explored bismuth resources has not allowed use of lead-bismuth coolant (LBC) in the large scale NP. However, just recently the specialized MINATOM enterprises – OAO “Atomredmedzoloto” and VNIPI of industrial technology – have carried out technical and economical investigations into an opportunity to organize large scale bismuth production in Russia and estimations of bismuth sources in the Commonwealth of Independent States (CIS). On the basis of the explored bismuth mines of the only Chita region in Russia, it is possible to produce bismuth in quantities sufficient enough to put n operation ~ 70 GWe of NPPs with LBC cooled FRs. In addition, there are large bismuth sources in the North Caucasus. It is possible to put in operation ~ 300 GWe by using the bismuth mines of Kazakhstan.

Japanese explorers have determined the world's available bismuth sources to be ~ 5 million tons [2].

It should be highlighted that in compliance with a general geological and economical law, the quantity of the mineral raw ore increases as the square of the cost that the consumer would be ready to pay.

At existing costs of bismuth in the world its contribution to the capital costs of constructing the large NPP on the basis of considered FRs is ~ 1 %. For that reason, in practice the real technical and economical parameters of the NPP will not be noticeably worth even in case of the bismuth cost increases several times.

In the future when cheap bismuth sources have been expired, it will be possible to change over to lead-bismuth alloy of a non-eutectic composition with reduced bismuth content and the higher boiling point. For example, when bismuth content in the alloy is reduced 5.5 times, the melting point is increasing from 125 to 250 °C that is considerably low than a melting point of pure lead.

This report is considering the NPT based on using small power lead-bismuth cooled FRs SVBR-75/100 (**L**ead-**B**ismuth **F**ast **R**eactor of equivalent electric power **75 … 100** MWe depending on the steam parameters). In the nearest 15 … 20 years it can be implemented both in developed and developing countries with meeting the most requirements to the Generation IV DOE reactor systems and International Project INPRO. Due to the higher technical and economical parameters of the NPP and the higher safety level [3], this technology can be considered as one of the possible ways of gradual replacement of the current NPT based on using the light water reactors (LWR).

An effect of improving the economic parameters of the NPPs based on RIs SVBR-75/100 is achieved due to lack of many safety systems necessary for the NPPs with LWRs, which make NPPs of this type considerably more expensive.

As it can be seen further, at the minimal starting costs of industrial mastering that NPT in the process of its evolutionary improvement, all these enable to implement gradual meeting of all requirements, which are peculiar to the mature phase of NP development.

2. BRIEF DESCRIPTION OF EXPERIENCE OF LBC USAGE

In the early 1950s, nearly at the same time the USA and the USSR launched their development programs on RIs for NSs. Both countries developed two types of RIs: pressurized water reactors and reactors cooled by liquid-metal coolants (LMC).

In the USA sodium was selected as LMC because its thermo-physical characteristics were better than those of LBC. The ground-based test facility-prototype of the RI and experimental NS “Seawolf” were constructed. However, operating experience revealed that option for the coolant, which was fire- and explosion-dangerous in contact with air and water, did not prove itself. After several RI accidents occurred at this NS it was decommissioned together with the compartment and replaced by a pressurized-water RI. R&D works on mastering LBC were also carried out in the USA. However, a selected approach of finding the solution to the problem of structural materials corrosion resistance, control and coolant quality maintenance (coolant technology) did not give any positive results, and the works were stopped.

In the USSR lead-bismuth eutectic alloy was selected as LMC in the very beginning. For fifteen years the certain organizations had been carrying out these works under IPPE scientific supervision. As a result, the problem was solved successfully, and it was verified by further many-year experience of RIs operating at the NSs. When operating the second generation RIs, there were no problems caused by structural materials corrosion in the primary circuit and violating the circuit purity standards [4].

The problem of ensuring radiation safety that was caused by forming polonium-210 was solved in the same way. During the whole period of operating LBC cooled RIs, including the primary circuit equipment's repair period and removal of spilled LBC, there were no cases of personnel's extra-irradiation over the permissible limits in terms of this radionuclide.

Altogether eight NSs with LBC cooled RIs were constructed. The first experimental NS of Project 645 had two reactors. Each of the other seven NSs of Project 705 (in terms of NATO – “Alpha”) had one reactor. Due to its speed parameters this NS was entered into Guinness Book of Records.

Besides, two full-scale ground reactor facilities-prototypes were constructed and operated in IPPE (Obninsk) and NITI (Sosnovy Bor). A total sum of operating time of the considered type RIs has been ~ 80 reactor-years [1]. The new nuclear power technology that has no analogues in the world has been demonstrated in industry. Currently the conditions for introducing this technology into the civilian nuclear power have been formed.

3. BASIC STATEMENTS OF RI SVBR-75/100 CONCEPT

RI SVBR-75/100 was designed mainly in compliance with a conservative approach. This approach allows: without exceeding the limits of the experimentally tested mode parameters of the primary and secondary circuits, to use to the maximal extent the already mastered fuel and structural materials and verified in practice the principal technical solutions to the equipment components and RI scheme.

This approach ensures a high extent of succession of the RI SVBR-75/100 technical solutions, first of all, the technical solutions of LBC cooled NSs' RIs that has been favoured by nearness of their scale factors. Adhering to this approach reduces the execution terms, R&D scopes and costs, investment risk, ensures reliability of the RI and its operation safety. These factors make it possible to avoid the errors typical of an initial stage of mastering the innovative NPT.

Use of the conservative approach does not mean that the new technical solutions should not be used and an evolutionary way of NP development should be only followed. This would cause stagnation and hindrance of the scientific and technical progress. However, use of verified in practice technical solutions ensures the applicable technical and economical parameters of the NPP [3]. For that reason, the new, perspective technical solutions that considerably improve the parameters of the RI will be used when changing over to the next generation of the given type RIs after carrying out the necessary R&D.

Expediency of this approach has resulted from the analysis of the technique development history. It shows that for successful introduction of new technologies, the share of new technical solutions in complicated systems should not be too high. Ignorance of this fact can result into considerable delay of start launching, unnecessary over-expenditure of materials and financing. For that reason, when RI SVBR-75/100 was being developed, priority was given to the already developed technical solutions even if they did not ensured achievement of the highest technical and economical parameters.

With due account of all mentioned, the following basic approaches and technical solutions have been realized in the RI SVBR-75/100 design:

(i) A monoblock (integral) design of a pool type is used for the primary circuit equipment. Valves and LBC pipelines are completely eliminated;
(ii) A two-circuit scheme of heat removal is used;
(iii) The levels of coolants' natural circulation (NC) in the heat-removal circuits are sufficient enough to ensure reactor's heat decay removal without dangerous over-heating of the core;

(iv) A reactor monoblock with a safeguard vessel is installed and fixed in the tank of the passive heat removal system (PHRS). The tank is filled with water and also performs the neutron protection function;
(v) A wrappless design of the fuel sub-assemblies (FSA) is used. This ensures high cross heat-mass-exchange in the core and eliminates unallowable over-heating of fuel elements at large blockages of flow rate at the core inlet;
(vi) A steam-generator (SG) operating in compliance with a multiple NC scheme and producing saturated steam is used. This ensures the best lifetime and operating parameters, e.g. reliable RI operation at any power levels, simplicity of maintaining LBC in a liquid state at low power levels (including the mode of heat decay removal via the SG);
(vii) A slow-rotating gas-tight uncontrolled electric engine of the main circulation pump (MCP), which power does not exceed 500 kW, is used. This eliminates the necessity to seal the rotating shafts, enables to use the ball-bearings with greasing and provides the necessary against-cavitation condition at the suction of the MCP impeller due to coolant column's hydrostatic pressure;
(viii) The RI equipment can be repaired or replaced;
(ix) On ending the lifetime, refuelling can be performed at once, FSA-by-FSA;
(x) It is possible to use different kinds of fuel (UO_2, MOX fuel with weapon or reactor Pu, TRUOX fuel, nitride fuel) without changing the reactor design and at meeting the safety requirements.

With due account of the relatively high cost of LBC, there have been developed the measures reducing the specific mass of LBC in the RI.

The summarized analysis of experience of developing RIs of different power capacities [5] has revealed the LBC specific mass decreases at reducing the RI nominal power.

Along with this, reducing the LBC specific mass is limited. It is caused by the fact that at small dimensions of the core, it is impossible to provide core breeding ratio (CBR) ≥ 1. Computations have revealed that an optimal diameter of the core should be not less than $1600 \div 1700$ mm at height 900 mm. These core dimensions make it possible to achieve equivalent electric power of the reactor ~ 100 MWe. In this case, CBR $\cong 1$ is provided not only for the mixed nitride fuel but also for the less dense but well mastered MOX fuel. This point can be carried out if the volumetric fuel fraction is not lower than $55 \div 60$ %.

Reduction of the LBC specific mass in small-sized FRs, for which the core power density is several times less than that of sodium cooled FRs, is also achieved by elimination of the in-reactor storage of spent nuclear fuel (SNF) and in-reactor refuelling mechanisms (rotating plugs, etc.).

In this case, refuelling is performed once during the core lifetime. For that purpose, a special refuelling equipment is used, it is also used for refuelling all reactors of the power unit. The refuelling technology is similar to that of LBC cooled NSs' RIs.

Another way of reducing the LBC specific mass is increasing its average velocity in the RI and diminishing the length of the LBC circulation circuit. However, this way has its own constraints caused by the necessity to provide the safety requirements.

The first requirement is caused by the necessity to provide the power level of the reactor with naturally circulating LBC to be not less than 5 % of N_{nom}. This makes it possible to eliminate dangerous temperature increase in case of shutting down the MCPs.

The second requirement is caused by the necessity to provide an effective separation of steam bubbles from LBC with steam surfacing to the LBC free levels in case of an accident with leaking SG tubes. This is necessary for elimination of steam ingress into the core and impermissible pressure increase in the monoblock vessel.

The necessity to satisfy the highlighted requirements resulted into development of the LBC circulation scheme in which core hydraulic resistance equals to 90 % of the total hydraulic resistance of the primary circuit and hydraulic resistance of the SGs, in which LBC flow rate is much less, only equals to 10 %.

With due account of the highlighted requirements, the specific mass of bismuth in RI SVBR-75/100 is ~ 1100 t/GWe.

It should be highlighted that the low values (25 … 30 %) of the LBC volumetric fraction in the core ("tight" lattice of fuel elements) and LBC specific mass do not deteriorate the safety parameters of RIs SVBR-75/100 in cases of shutting down the MCP and leaking SG tubes (as computations have revealed) but in the case of unauthorized insertion of positive reactivity as well. The latter is caused by a sufficiently high negative feedback being typical of small power reactors and a low time of delaying its temperature component at the LBC core inlet (extending of the lower core-plate) coupled with sufficient heat-accumulation ability of the monoblock.

The following have been provided at the selected power level (100 MWe):

— The lifetime duration is ~ 53000 eff. hours if mastered oxide uranium fuel is used (CBR = 0.87);
— CBR ≥ 1 if MOX fuel is used, the reactor operates in the closed fuel cycle in the mode of fuel self-providing;
— CBR ≥ 1 if mixed nitride fuel is used, the reactor operates in the mode of fuel self-providing at a burn-up reactivity margin being less than β_{eff} or in the mode of extended breeding with CBR = 1.13 at a plutonium doubling time being ~ 45 years;
— A burn-up reactivity margin is less than β_{eff}, the lifetime duration is ~ 80000 eff. hours in case of using uranium nitride fuel;
— Reactor's heat decay removal is entirely passive, heat is removed through the monoblock vessel to the PHRS tank;
— Complete plant fabrication of the reactor monoblock, RIs are produced in large quantities that improve the quality of works and reduce the cost;
— The reactor monoblock can be transported by railway, truck or marine transport (with fuel in a nuclear and radiation-safe state due to LBC "freezing" in the monoblock vessel that also meets non-proliferation requirements);
— The term of constructing the NPP unit can be considerably reduced as modules are delivered in high plant readiness and the assembling scopes are sharply reduced. (This improves the terms of receiving the NPP construction credits and reduces the period of capital investments recoupment);
— The NPP unit in which the RIs have been replaced by the new ones can be renovated in 50 … 60 years. This postpones the necessity to construct the replacing power capacities to 50 years;
— The cost of decommissioning the unit can be considerably reduced as after removing the monoblock, no radioactive materials remain in the main reactor building;

The NPP units with LWRs which RIs have exhausted their reactor lifetime can be renovated by installing the necessary number of RI SVBR-75/100 in the empty SG and MCP rooms.

The basic parameters of RI SVBR-75/100, a longitudinal section of the reactor monoblock and reactor compartment are cited in Table I, Fig. 1, Fig. 2.

Table I. Basic parameters of RI SVBR-75/100

	Name and Dimensions of the Parameter	Value
1	Heat power (nominal), (MW)	280
2	Steam-production, (t/h)	580
3	Pressure of generated saturated steam, (MPa)	9.5
4	Feed water temperature, (°C)	240.9
5	Primary circuit coolant's flow rate, (kg/s)	11760
6	*Primary circuit coolant's temperature, outlet/inlet, (℃)*	482/320
7	Core dimensions: diameter × height, (m)	1.645 × 0.9
8	The number of fuel elements	12114
9	The number of CPS rods	37
10	Average power density of the core, (kW/dm$^{3)}$	146
11	Average linear load of the fuel element, (kW/m)	~24.3
12	The time interval between refuellings, (years)	~8
13	Uranium fuel (UO_2) load: mass, (kg)/enrichment, (%)	9144/16.1
14	The number of SG modules	2 × 6
15	The number of MCPs	2
16	Power and head of the MCP, (kW/MPa)	450/0.55
17	The core lifetime, (eff. hours)	53000
18	LBC volume in the primary circuit, (m^3)	18
19	Dimensions of the reactor monoblock: diameter × height, (m)	4.53×7.55

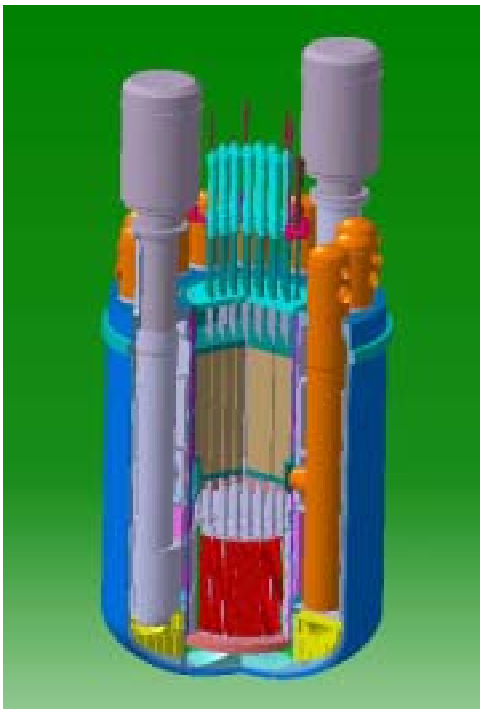

FIG. 1. Reactor monobloc.

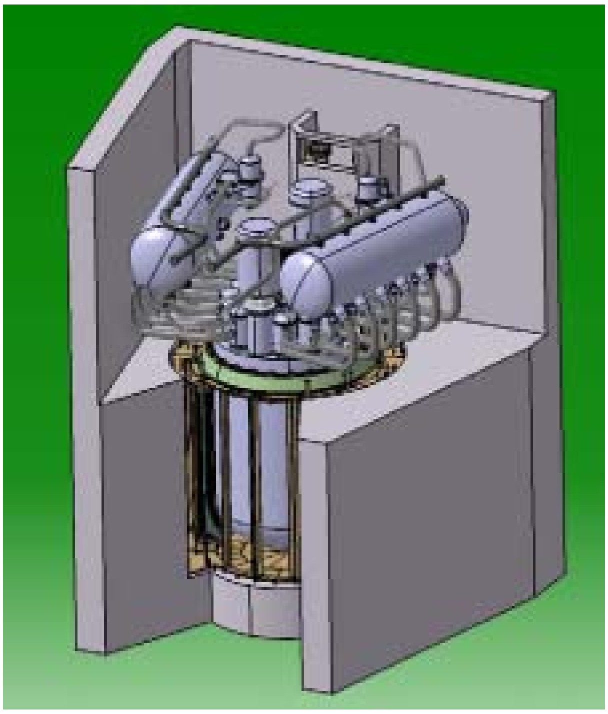

FIG. 2. Reactor compartment.

4. SAFETY PROVIDING CONCEPT

Lead and bismuth natural properties, physical features of FRs coupled with an integral (monoblock) design of the primary circuit equipment make it possible to eliminate deterministically an opportunity of the certain severe accidents.

High boiling point of coolant enhances reliability of heat removal from the core and safety due to lack of the heat removal crisis phenomenon and being coupled with a safe-guard vessel eliminates the accidents of the LOCA type.

Low pressure in the primary circuit enables to reduce the thickness of the monoblock vessel walls and reduce the limitations imposed on the temperature change rate in compliance with the thermo-cycling strength conditions.

LBC reacts with water and air very slightly. Development of the accident processes caused by primary circuit's tightness failure and SG intercircuit leaks occurs without hydrogen release and any exothermic reactions. There are no materials within the core and RI that release hydrogen as a result of thermal and radiation effects and chemical reactions with coolant. Therefore, the likelihood of chemical explosions and fires as internal events is virtually eliminated.

In the case of failure of all active cool-down systems and total blacking out the unit, elimination of core melting caused by residual heat release and keeping the monoblock vessel intact are ensured by an entirely passive way due to heat accumulation in the in-vessel structures and coolant and heat removal to the PHRS water tank through the monoblock vessel with further water evaporation. The “grace” period is about five days' time. A scheme of heat removal to the PHRS tank is shown in Figure 3 and Figure 4 shows how the maximal temperature of the fuel element's cladding and the water level in the PHRS tank depend on time.

Core melting is also eliminated at postulated LBC “freezing” in the SG. In this case, NC of LBC with a flow rate being ~1 % of the nominal one is performed over the continuously operated by-pass circuit past the SG from the central buffer chamber to the peripheral one via the holes in the shells, which have been provided for this purpose.

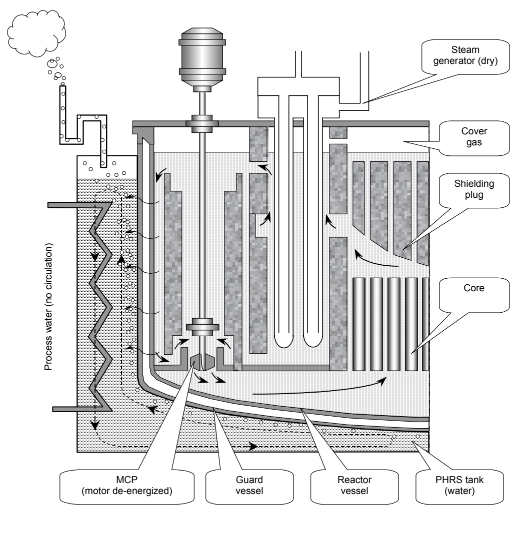

FIG. 3. Heat removal to the PHRS tank.

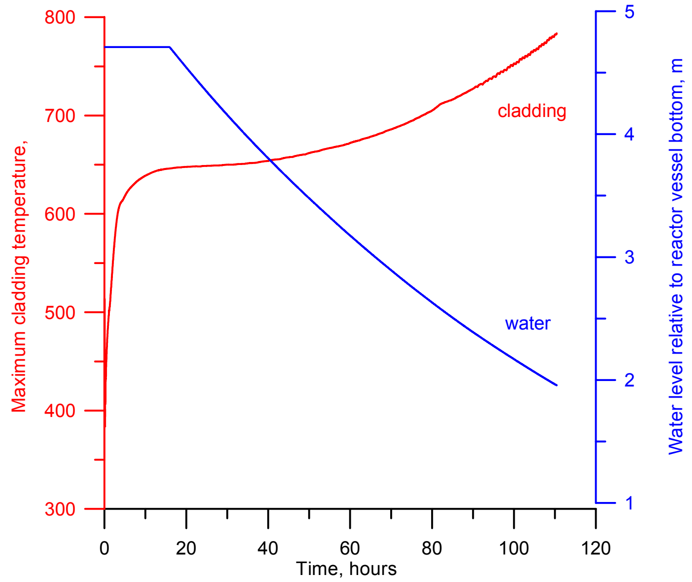

FIG. 4. Fuel element's cladding and the water level in the PHRS tank depend on time.

In case of unauthorized insertion of positive reactivity at postulated failure of all emergency protection (EP) drivers, elimination of prompt neutron reactor runaway is ensured by a special algorithm of compensating rods control, which is the part of the automatic control system. In this case, when the reactor operates at nominal power, during a certain time (~ 4 months) a reactivity margin controlled by an operator is much less than $\beta_{eff.}$

Besides, an efficiency of each rod is much less than β_{eff}, a rate of moving the absorbing rods extracted gradually is technically limited. For that reason, the inserted positive reactivity has time for being compensated by negative feedbacks without dangerous increase of the core temperature.

In the case of EP system failure caused by the external events not specified in the regulatory documents (for example, damage of all servo-drivers), there are fusible locks connecting a rod with a driver bar. When the coolant temperature exceeds 700 °C, EP rods that are installed in the "dry" channels are separated from the bars and drop into the core due to their gravity.

For considered fuel loads, the total void reactivity effect of the reactor is negative and the local positive void reactivity effect is less than $\beta_{eff.}$ and cannot be realized due to the coolant's very high boiling point and lack of the opportunity for gas or steam bubbles to arise in great quantities.

Elimination of water or steam penetration into the core caused by a large SG leak and consequent overpressurization of the monoblock vessel designed to be resistant against the maximum possible pressure under this condition are ensured by the coolant's circulation scheme. This scheme provides that steam bubbles are thrown out on the free coolant level by the moving up LBC flow. Then steam goes to the gas system condensers. In the event of their postulated failure, steam goes to the bubbler (PHRS tank) through the bursting membranes (see Fig. 5).

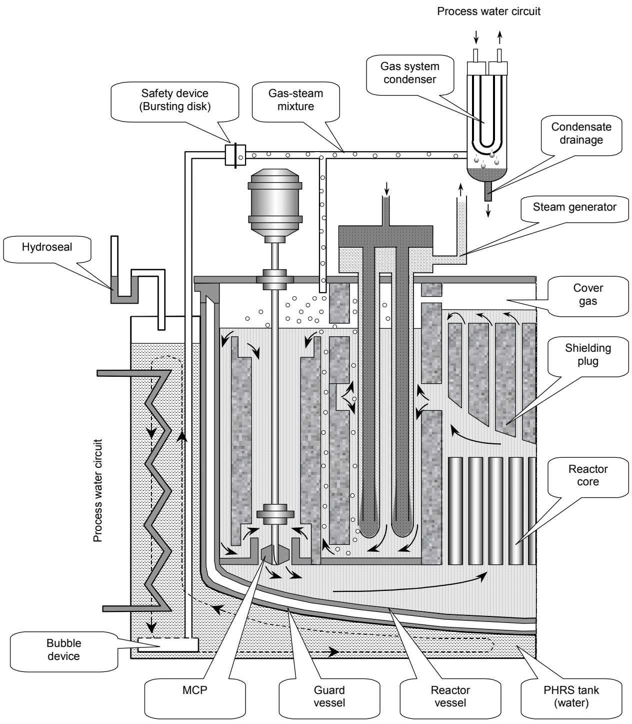

FIG. 5. Heat removal to the PHRS tank in case of failure.

Carried out studies have revealed that no equipment failures, personnel's errors or their combinations may cause core melting. Negative reactivity feedbacks ensure power decrease down to the value that does not cause core damage even in case of failure of all reactor shutdown systems and total blacking out.

The additional barriers of the safety providing system are the separate concrete cells of the RI (confinements) that restrict radioactivity release into the central reactor hall, and a protection shell of the central hall covering all RIs (containment) purposed to protect the reactor against external impacts. In an event of accident tightness failure of the primary circuit, high pressure radioactive exhausts (which can happen in LWRs) do not occur in LBC cooled RIs. For that reason, there is no need to design a containment of the unit and RI compartment to be resistant against high excess internal pressure. There is also no need to design a double containment with a water cooling system and a corium catch.

An extremely simple design of RI SVBR-75/100, lack of plant safety systems caused by developed inherent safety properties of the RI, that made it possible to couple the functions of RI safety systems with those of normal operating systems, sharply reduce the probability of personnel's errors. The consequences of any personnel's errors and their combinations do not affect the safety but only result in economical losses and the necessity to carry out unscheduled repair works.

RI safety does not depend on the equipment and systems of the turbine-installation. Major safety viable systems operate passively being independent on either right or wrong personnel's actions.

It should be highlighted that there are no valves or mechanical devices in the RI safety systems, which may cause failure of their operation in case of failure or switching off the valves or mechanical devices that may be caused by someone's error or malicious actions or over-standard external impacts.

For that reason, RI safety systems' action has been assured by:

— Melting the locks of the EP rods and their free fall down to the core;
— LBC natural circulation, heat transfer via the main and safe-guard vessels, air convection in the gap and heat irradiation, water boiling in the PHRS tank in the modes of emergency heat decay removal;
— Rupture of the safety membrane that protects the monoblock from excess pressure at large SG leaks and failure of the gas system's steam condensers.

As computations have revealed, an extremely high safety potential typical of the considered RI is characterized by the following: even in an event of the postulated combination of such initial events as containment destruction, damage of the RI compartment overlapping and primary circuit gas system's serious failure with direct contact of the LBC surface with atmospheric air in the monoblock vessel that is possible in the case of terror attacks, neither reactor runaway, nor explosion, nor fire occurs, and the radioactivity release is less than that requiring the population evacuation.

Obtained results enable to conclude that the safety level of SVBR-75/100 reactors is higher than that of LWRs and sodium cooled FRs. It can be practically demonstrated at the stage of experimental operation of RI SVBR-75/100 with controlled simulation of different designed initial events and their combinations.

5. CONCEPT OF THE MODULAR NPP BASED ON RI SVBR-75/100

It is highlighted in [6] that “Modular plant fabrication of nuclear power systems and their assembling on the site will replace the existing expensive construction methods”. Economical advantages of modular principle of constructing the NPP are also highlighted in [7]: “Measures on reducing the construction terms much affect the total capital costs especially at high record rates because in the course of construction, the credit payment may reach 25 percent and more of the total scope of investments. Modular production that makes it possible to fabricate and assemble the units at the plant but not on the site reduces the construction term and, consequently, expenses on the credit payment during the construction period”.

Reduction of the investment cycle of constructing the NPP, that has been ensured by a modular structure of the NPP and delivery of ready fabricated modules, is extremely viable for the technical and economical parameters of the NPP to approach those of steam-gas HPPs with short investment cycles [8].

For developed countries, which power systems have high-voltage electric transfer lines with high transmission, it will be economically effective to use large modular power units. Maximal possible capacity of the modular type power-unit will not be restricted by maximal possible reactor capacity.

In case the large power modular power-unit is equipped with one turbine installation and at the existing technical level of turbine-constructing factories in Russia, the power-unit's capacity can be taken as 1600 … 1800 MWe. SSC RF IPPE, FGUP EDO “Gidropress”, FGUP “Atomenergoproekt” have developed a conceptual design of the two-unit NPP, which power unit includes the nuclear steam-supply system (NSSS) consisting of 16

RIs SVBR-75/100 (reactor modules) and one turbine-installation of 1600 MWe [3]. This allows to compare correctly the technical and economical parameters of that NPP to those of the NPP based on RI VVER-1500.

When NPP unit's capacity was selected, it was taken into account that specific capital costs of the reactor compartment ("nuclear island") would decrease at increasing the unit's capacity. It is caused by the fact that at increasing the number of modules in the NSSS, the cost of the equipment and providing systems installed beyond the RI compartments increases slightly. For that reason, their contribution to the specific capital costs of the reactor compartment will decrease.

Such systems and equipment include the refuelling equipment, coolant's in-taking equipment, equipment for coolant's transferring to the monoblocks at initial filling, etc. So, the specific capital cost of constructions necessary for installing these systems will be reduced correspondingly.

A modular principle of the NPP design is the most economically effective for reactors, in which the inherent safety properties against severe accidents have been realized to the maximal possible extent. First of all, this means the accidents with coolant's loss such as LOCA. To overcome these accidents, the LWRs need a lot of safety systems that are not necessary for RIs SVBR-75/100. This considerably reduces the construction scopes of the reactor compartment.

Control of the modular NSSS is carried out by an operator who uses the common power master unit. If there is any fault in the certain RI, it is automatically removed out of operation and can be cooled down autonomously with the turbine-installation systems.

A simple scheme of the RIs and similarity of their types allow to reduce the number of the operation and maintenance personnel at the modular NPP unit as compared with that at the NPP unit with one large-power RI with lots of safety systems including protection systems of localizing the accidents, control and providing systems. For example, the safety systems of the AP-1000 reactor have 184 pumps, 1400 driver valves, 40 km of the pipelines and cables [9].

A modular design of the NSSS power unit makes it possible to provide LF to be not less than 90 % under long reactor operation without refuelling. When each RI is shut down for refuelling, power unit's power reduces slightly.

Once-moment sequential refuellings of each RI included into the NSSS are equivalent to the mode of partial refuellings of the large-power reactor (1600 MWe) at annual refuellings of ~ 1/8 share of fuel each year). Duration and periodicity of scheduled maintenance and repair works are determined by requirements to the turbine-installation equipment.

Licensing of constructing the modular type large power power-unit will be much simplified in the case of constructing one RI or the small power modular power-unit which RI has been certified. Small power of the RI determines a comparatively low cost of its construction.

A plan and a longitudinal section of the SVBR-1600 reactor compartment's main building with the NSSS are shown in Fig. 6. The basic technical and economical parameters of the two-unit NPP based on RI SVBR-75/100 in comparison with those of the two-unit NPPs with RI VVER-1500, RI VVER-1000 (V-392), RI BN-1800 and HPP with ten steam-gas units PGU-325 are summarized in Table II [3].

Table II. Comparable parameters of different power plants

Name and Dimensions of the Parameter	NPP with RI SVBR-75/100	NPP with RI VVER-1500	NPP with RI BN-1800 [10]	NPP with RI VVER-1000	HPP with PGU-325
1. Set up power of the power-unit, (MWe)	1625	1479	1780	1068	325
2. The number of the units at the plant	2	2	2	2	10
3. Electric power necessary for plant's own needs, (%)	4.5	5.7	4.6	6.43	4.5
4. Efficiency of the net plant (power unit), (%)	34.6	33.3	43.6	33.3	44.4
5. Specific capital investments in the industrial construction of the plant, ($/kW)	661.5	749.8	783.4	819.3	600
6. Design cost of produced electricity, (cent/kW·h)	1.46	1.85	1.56	2.02	1.75

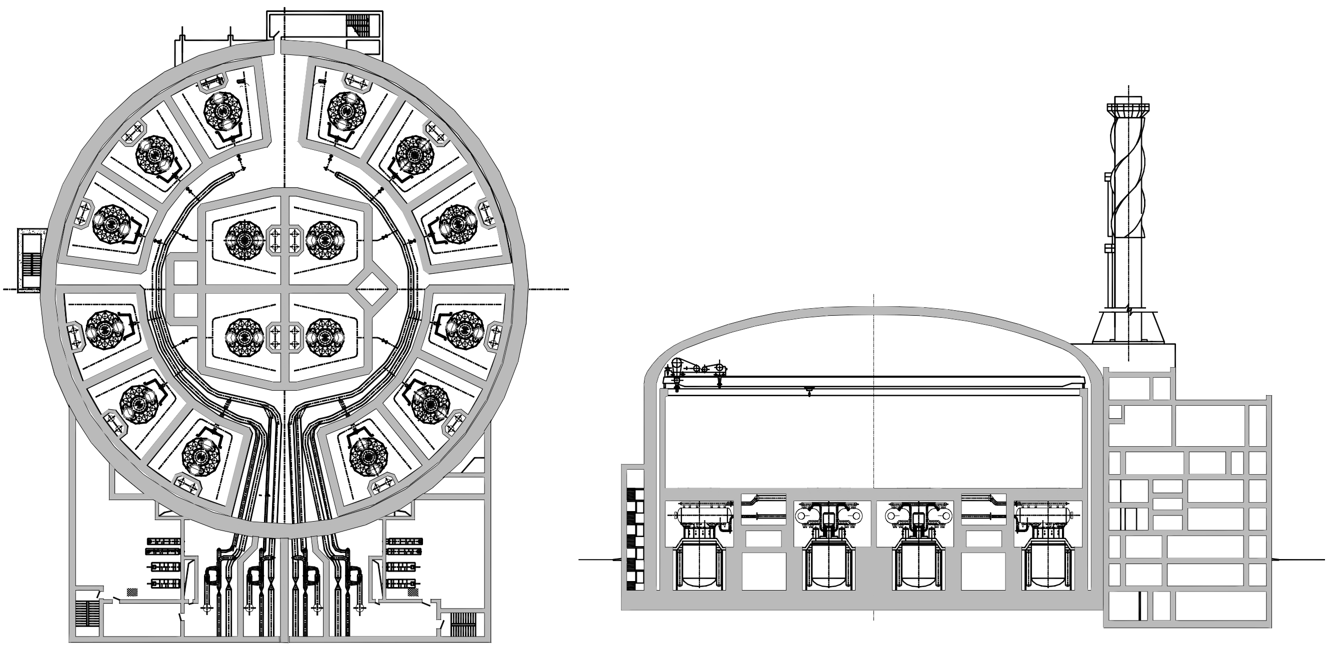

FIG. 6. A plan and a longitudinal section of the SVBR-1600 reactor compartment's main building with the NSSS

The results of technical and economical computations have revealed that in compliance with the data obtained at the conceptual design stage, the technical and economical parameters of the NPP with two 1600 MWe units, each based on the SVBR-75/100 type RI, are better than those of the NPP based on the large power LWRs and than those of the HPP with ten units PGU-325 operating by using natural gas. The term of constructing this NPP can be ~ 3.5 years.

However, it should be highlighted that reliability of the sited economical parameters of LWRs, which have been developing during several generations, is higher than those of the NPPs with RIs SVBR-75/100 because they have not had experience of practical realization. For that reason, the costs of the capital investments in the industrial construction of the NPP based on RIs SVBR-75/100 has an additional margin of 17 % over the standard one (60 % of the RI equipment cost). Besides, this NPP project that is actually the first generation design based on the conservative approach has a great potential for its development.

In the future, after carrying out corresponding R&D, the following technical solutions would improve the RI design and considerably improve the technical and economical parameters:

Use of the one-through SG producing super-heated steam that makes it possible to increase an efficiency of the thermo-dynamic cycle and increase electric power by 10 … 15 %;

Increasing the LBC temperature at the reactor outlet at increasing the maximal temperature of the fuel elements' cladding from 600 to 650 °C that provides increasing the reactor heat power by 15 … 20 % without changing its design.

Realization of these measures will make it possible to reduce the specific capital costs of plant's industrial construction to 560 $/kW and probably to a lower value.

6. FUEL CYCLE

Due to the low current costs of uranium and its enrichment, use of oxide uranium fuel with postponed reprocessing and SNF storing on the NPP site is economically justified for RI SVBR-75/100. Duration of this stage depends on the available resources of cheap uranium and NP scales. In compliance with the [11] data, in Russia an estimated term of expiring the cheap uranium resources will be 70 years at an average level of NPPs' total power being 45 GWe and in the world this value will be 40 years at an average level of NPPs' total power being 750 GWe.

However, because the increase of natural gas costs will overtake the increase of uranium costs, competitiveness of the NP will be assured even at higher costs of uranium caused by the fuel component's lower share in the electricity cost of the NPPs as compared with that of the HPPs.

At this stage the major way of improving the economic parameters of the fuel cycle will be increasing the lifetime duration (fuel burn-up depth) as experience in the core elements operation ability is gained and use of vibro-dense oxide fuel allowing deeper burn-up.

Further, an economically expedient will be the stage at which the own SNF will be reprocessed, NFC will be uranium closed (at adding enriched uranium into the NFC), plutonium, MA, fission fractions will be extracted and then stored.

Duration of the uranium stage can increase when changing over to the denser nitride fuel. At the same time, it could be expedient to only use uranium nitride for export in order to reduce the risk of unauthorized fissile materials proliferation at expanding the refuelling interval to only 15 years.

Actually, in the future it will be necessary to change over to the entirely closed NFC. The time period required for this change will be determined by appearing the developed in an industrial scale technology of SNF reprocessing that will be acceptable from the standpoint of RAW minimization and fissile materials non-proliferation. The existing technology of radiochemical reprocessing SNF do not meet these requirements. Besides, it will be only economically justified under stable operation of a large radiochemical factory (i.e. at reprocessing scales being 1000 … 1500 t of SNF per year that corresponds to the total level of NPP set up power being ~60 … 90 GWe).

One of the economically expedient variants of changing over to the entirely closed NFC that meets the necessary requirements is a technology based on using the “dry” methods of reprocessing SNF and a vibro-pack technology when the fuel elements are fabricated.

SSC RF-NIIAR have carried out the researches revealing that construction of power capacities on reprocessing SNF of SVBR-75/100 reactors and fresh FSA fabricating increases the specific capital costs of constructing the NPP by not more than 10 … 15 % (about $ 76 /kW of set up power). And it has been presumed that reprocessing is performed on the basis of the pyro-chemical processes in the chloride melts and the reprocessing rate is 120 t of heavy metal per year (the highlighted reprocessing rate corresponds to the total NPP power on the basis of RIs SVBR-75/100 being ~ 12 GWe).

Change over to the closed NFC for the SVBR-75/100 reactors will have the lower cost if for fabricating the first fuel load from MOX fuel we use plutonium that has not been extracted from LWRs' SNF but has been extracted from the own SNF of uranium loads due to considerably lower scopes of reprocessing in terms of 1 t of plutonium. The quantity of plutonium extracted from SNF of three uranium cores is enough for fabricating one core from MOX fuel.

When reactors SVBR-75/100 operate in the closed NFC, economically effective use of LWRs' SNF as make up fuel without separation of uranium, plutonium, MA, and fission fractions instead of waste pile uranium is possible (similarly to the DUPIC-technology for the CANDU-reactors). That is, instead of reprocessing SNF of thermal reactors (both from the VVER and RBMK reactors) for the purpose to only extract 1 % of plutonium, after long storing during ~50 years this kind of SNF will be step by step utilized in the FR.

Due to the fact that the fraction of LWRs' SNF in fresh fuel of SVBR-75/100 operating in the closed NFC is ~ 10 … 12 % and the plutonium fraction in LWRs' SNF does not exceed 1 %, influence of the plutonium isotopic vector in LWRs' SNF on the isotopic vector of fresh fuel is negligible for SVBR-75/100. Therefore, RI SVBR-75/100 makes it possible to develop a principally new strategy of the closed NFC that does not require expensive reprocessing SNF of thermal reactors for the purpose to extract plutonium for FRs' fuel supplying.

Flexibility of RI SVBR-75/100 relative to the fuel cycle technologies that is realized in compliance with a principle: “To operate using the type of fuel that is the most effective” makes it possible to postpone a task of constructing a specialized enterprise to several decades after the first unit of the NPP with such reactors has been put in operation. For example, after introduction of about 10 GWe of power capacities on the basis of RIs SVBR-75/100 and repaying the costs of NPP construction, a certain share of profit could be spent on developing the industry on SNF reprocessing and fabricating FSA from MOX fuel.

After launching that factory, the cost of the core would be only determined by current operating costs of SNF reprocessing and FSA fabricating. If the SSC RF-NIIAR designs are used as a basis of that complex, contribution of fuel costs to the cost of the SVBR-75/100 core would be even less than that of the basic variant using oxide uranium fuel. This will make it possible to considerably improve the NPP competitiveness. This approach to construction of power capacities on SNF reprocessing and FSA fabricating presumes the owner of the NPP units is also the owner of the fuel cycle enterprise.

Along with this, on considering the prospects of economically substantiated closing the NFC of FRs caused by lack of uranium, some experts presume it might be required in 50 … 100 years [12].

In this case it has been accounted that the cost of natural uranium is a very small fraction in the cost of electricity production. Even the opportunity of extracting uranium from the sea water (that seems exotic today) has been assessed and that cost will be 500 $ per kg.

7. LONG-LIVED RAW MANAGEMENT AND ENVIRONMENTAL IMPACT

In the course of NPP operation, liquid RAW is produced in very low quantities. This fact has been verified by experience of operating LBC cooled NSs' RIs.

The NPP design provides an installation for concentrating and solidifying the low quantities of liquid RAW.

After expiring the RI lifetime, the radioactive LBC can be many times recycled in the new RIs. In 1000 years of irradiation, slight residual long-lived radioactivity of LBC caused by Bi-208 and Bi-210m radionuclides will be lower than natural radioactivity of the uranium ore (in terms of U_3O_8). It will be only important at the final stage of NP functioning.

In this connection, LBC in the form of solid radioactive waste being disposed in the deep geological formations will not disturb the natural radioactivity equilibrium. Low chemical activity of lead and bismuth rules out radioactivity release into the biosphere. Therefore, the radio-ecological consequences of this disposal will be of no risk for the population of the next generations.

There is a similar problem for the LWRs as long-lived radionuclide zirconium-93 is forming in the fuel elements' zirconium claddings and channels.

The quantity of tritium release into the environment due to unavoidable water losses in the RI secondary circuit does not exceed ~ 50 TBq/GWe*yr that is within the limits of normalized release of tritium with liquid wastes into the environment of the world's operating NPPs.

Radioactivity release into the environment out of unloaded SNF is eliminated by a multi-barrier shielding against activity release out of the spent FSA. Being unloaded out of the reactor they are installed into the steel capsules filled with liquid lead. After lead solidifying four barriers are formed in the capsules: the fuel matrix, fuel element cladding, solidified lead, capsule vessel.

When operating in the closed NFC, fission products management does not presume their transmutation because of the low efficiency of the process.

Taking into account that the half life of majority of fission products does not exceed 30 years (except for technetium-99, iodine-129, cesium-135, and some others), it is supposed that after vitrifying they are placed into the "dry" control storage for about 300 years of long storing. After that storing, their activity will be determined by long-lived nuclides of technetium, iodine and cesium.

It is proposed to dispose these vitrified fission products in the deep geological formations with providing a multi-barrier shielding. (Instead of vitrifying a "sinrock"-technology may be used after verifying its advantages.) That method of fission products management rules out radioactivity release into the environment.

Management of transuranium (TRU) elements presumes that their release beyond the fuel cycle will be eliminated (except for very low losses at the stage of RAW chemical reprocessing) as they are well fissionable in a hard neutron spectrum of FRs and their concentration achieves a saturation condition very quickly.

To estimate the environmental impact caused by the NFC of SVBR-75/100, a value of specific radiotoxicity of transuranium elements (neptunium, plutonium, americium and curium) and long-lived fission products (technetium-99, iodine-129 and cesium-135) as a function of produced electricity was taken as a criterion.

The radiotoxicity standard was adopted as a volume of water necessary for diluting a built-up quantity of radionuclides to decrease its concentration down to the level meeting sanitary requirements to the drinking water in terms of specific radioactivity. Specific radiotoxicity is determined as SNF radiotoxicity divided by produced energy.

The following assumptions were made to evaluate radiotoxicity:

— MOX fuel with plutonium extracted from LWRs' SNF was used as a first load of the reactor;
— At the end of each lifetime and three-year cooling, SNF was reprocessed;
— Radiotoxicity of the main bulk of fission products with half-lives being less than 30 years was not accounted as after 300 years of cooling their radiotoxicity would be very low;
— Curium was extracted and transported to the temporary storage (repository) for 100-150-year cooling. After cooling, all radioactive isotopes of curium (except for curium-245) were transformed into plutonium isotopes. Then this isotopic mixture was transported back to the reactor for its further incineration [13];
— Mixture of plutonium, neptunium and americium with the rest uranium and necessary addition of depleted (waste pile) uranium was used for fabricating the fuel load for the next lifetime.

Figure 7 presents SNF long-lived specific radiotoxicity as a function of produced energy for reactor SVBR-75/100 within the NFC. Specific radiotoxicity of technetium-99, iodine-129 and cesium-135 in the final disposal is 0.014 km^3 /GWe*year that is nearly equal to that of natural uranium annually added to the fuel cycle in terms of GWe*year.

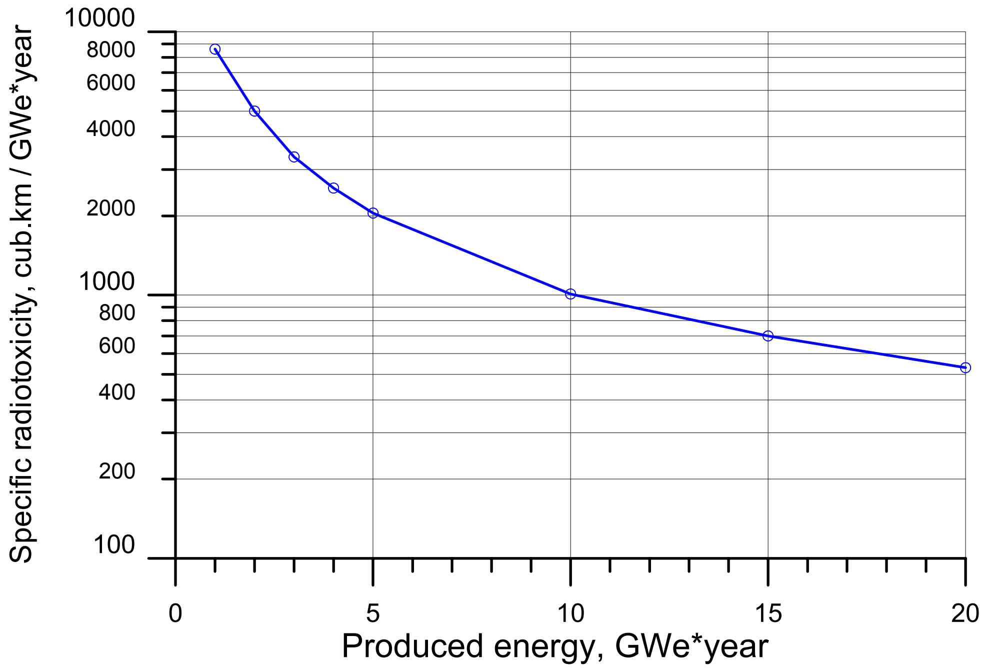

FIG. 7. SNF long-lived specific radiotoxicity.

The analysis of the obtained results shows the environmental-"friendly" effect of the NFC of SVBR-75/100 as specific radiotoxicity of long-lived RAW decreases at increasing the value of cumulative produced energy to the value of specific radiotoxicity of the extracted uranium ore. This is caused by the fact that the hard neutron spectrum in the reactor facilitates efficient incineration of both own MA, and MA built up in the LWRs.

8. TECHNOLOGICAL MAINTENANCE OF NON-PROLIFERATION

Non-proliferation of fissile materials means creating the conditions when inappropriate use of fissile materials is least attractive for potential distributors of the nuclear weapon.

It is evident that the problem of non-proliferation cannot be only solved by technological measures as despite the development of a new nuclear technology, there are the opportunities for illegal receiving the weapon materials and using the well-developed technologies of isotopic uranium separation and plutonium extraction out of spent fuel. For that reason, the complete solution to the problem of non-proliferation can be only achieved by coupling the technological and political measures.

Relationship of these measures will be different for nuclear and non-nuclear countries. During the recent decades all nuclear countries, which legally possessed the nuclear weapon, have solved this problem successfully using the measures of physical protection, accounting, control and safeguard. For that reason, the additional measures of technological maintenance of non-proliferation will be justified in case they do not reduce the NP competitiveness.

When using the NPP in developing countries, the additional measures of technological maintenance of NFM non-proliferation should be taken along with the political measures and international control.

In case of the export deliveries to the developing countries, the reactor design must eliminate access to the fuel. A reactor seller should keep the property rights to the reactor and core and provide necessary maintenance caused by periodical replacement of the core/reactor. A reactor module should be designed as a wholly replaceable unit after expiring the core lifetime (the goal is 15 years). This approach assures that the User-country should not possess the refuelling equipment, or the spent fuel repositories.

It is expedient to concentrate SNF reprocessing at the certain "closed" factories in nuclear or developed countries.

In the far future, when FRs are used world-wide, it would be possible to give up gradually the technologies of uranium enrichment and pure plutonium extraction for NP needs.

To reduce the risk of NFM unauthorized proliferation, the following measures realized at the different stages of the RI SVBR-75/100 lifetime including the NFC are considered below:

- The RI design eliminates uranium and thorium blankets that make it possible to accumulate fissile materials for the weapon purposes.
- Refuelling is performed very seldom (once a 7 … 10-year period) and can be inspected easily. Partial refuelling is impossible.
- Refuellling is only performed by using the special equipment kit that is not delivered to the User-country.
- At the stage of fabricating the initial fuel load by using oxide uranium fuel, NFM non-proliferation is ensured by the core design due to use of uranium with U-235 enrichment being less than 20 %. This satisfies the IAEA requirements and allows to use these reactors in non-nuclear countries.
- At the stage of SNF storing, proliferation resistance is ensured by the fact that built-up plutonium together with high radiotoxic fission products is in the SNF ("the spent fuel standard"). Therefore, the possibility to steal SNF is eliminated and the SNF movements can be easily inspected by gamma-radiation.
- At the stage of SNF reprocessing, proliferation resistance is ensured by the fact that during technological reprocessing, built-up plutonium along with built-up MA is separated from uranium at non-deep purification from fission products. Therefore, plutonium stealing is impeded, and its applicability for fabricating the explosion devices becomes insignificant.
- At the stage of fabricating and transporting the MOX fuel, proliferation resistance is ensured by the fact that during fabricating re-fabricated fuel, 2 % of fission products built-up in SNF and all MA remain in it. This requires remote management of that fuel, impedes its stealing and facilitates the inspection of its movements.
- This fuel can be delivered to any countries as fuel management is only possible by using the special large heavy equipment that facilitates the accounting and inspection of fresh fuel.
- Fuel transportation in the reactor monoblock with solidified LBC creates an additional technical barrier to the fuel thefts.
- The IAEA inspection is ensured at all stages of the NFC.
- The measures of physical protection and safeguard are used.

9. POSSIBLE AREAS OF USING RI SVBR-75/100

High technical and economical parameters of RI SVBR-75/100, ability of the reactor monoblocks to be transported by railway, inherent safety properties of the RIs make conditions for their multi-purpose usage when they have been produced in large quantities.

First of all, it is renovation of the NPP units with LWRs which RIs have expired their lifetime. They can be renovated by installing the necessary number of RIs SVBR-75/100 in the empty SG and MCP compartments.

Results of technical and economical researches into technical opportunity and economical expediency of renovating the 2-nd, 3-rd, and 4-th Novovoronezh NPP (NVNPP) units on the basis of RIs SVBR-75 have revealed that renovation reduces two times the specific capital costs as compared with construction of the new replacing power capacities [14].

A similar renovation technology can be used for almost all LWRs units. In this case, the capital costs saving will be $ 500 M per GWe (in Russian conditions) as compared with construction of the new replacing power capacities.

Experience gained by operating an industrial prototype of RIs SVBR-75/100 in conditions of the NVNPP will make it possible with a minimal investment risk to launch sequential renovation of the LWR units, which RIs have expired their lifetime and construction of modular NPPs with large power units in the countries, which power systems have high-voltage electric transfer lines with high transmission.

Taking into account the high extent of inherent safety of these RIs, it is expedient to use them for the heat supply needs.

In Russia: these are the regional NHPPs of 200 … 600 MWe, which are necessary to be located near the cities. The term of constructing the regional NHPPs and their total cost will be much less than those of large power NPPs. Their construction can be realized at the expense of the finance sources of the Russian Federation subjects, including the joint stock but this require for the legislation to be changed.

Abroad: these are the power-complexes designed for producing electricity, heat and water desalination. Carried out by IAEA marketing studies have revealed that in developing countries that have no powerful electricity-transfer lines, there is a vast market for small power reactors of ~ 100 MWe.

Export potentials can be realized by granting on lease the transportable reactor unit for steam-supplying these power-complexes. In this case, the Supplier keeps the property rights, the Consumer needn't develop and maintain the complex infrastructure of fuel management, the Supplier takes all the possible risks.

In this case, the requirements to fissile materials non-proliferation are ensured by using uranium enriched in low than 20 %, lack of refuellings in the User-country. For refuelling the reactor unit should be transported to the User-country (once a 10-year period) in a nuclear and radiation-safe state due to the “freezing” LBC and core in the monoblock vessel.

10. CONCLUSION

— The inherent safety properties of RI SVBR-75/100, economic competitiveness of the NPPs with their usage, opportunity to operate in the closed NFC in the fuel self-providing mode or with low breeding, opportunity both to burn own MA in the reactor and use LWR's SNF as make up fuel, providing technological maintenance of non-proliferation make it possible to consider the proposed reactor technology as one of the most perspective trends in NP development.
— RI SVBR-75/100 meeting most of the requirements to the Generation IV DOE reactor systems and IAEA Project INPRO can be proposed as one of the basic installations of the collaborative International Project.
— The obtained results have revealed the technical opportunity and economical expediency of using the RIs of the SVBR-75/100 type for finding the solution to the certain basic tasks of Russian NP both in the nearest and far future at minimal launching costs of industrial mastering. First of all, this refers to the opportunity of economical effective considerable extending the NPP units' lifetime (by 30 … 40 years) by renovating them.
— Use of the modular structure of the power-unit's NSSS makes credible an opportunity to change over in the future to the innovative technologies of a standard design of various capacity power units on the basis of the standard modules produced in quantities and a conveyer method of carrying out assembling works. This will make it possible to considerably reduce the terms of NPP construction and use a service base for technical maintenance of the reactor modules. The maintenance personnel will be also considerably reduced.
— Nuclear power based on the considered type NPPs can compete with heat power based on the modern steam-gas HPPs not only at the liberalized power market but at the investment market that will ensure the necessary pace of its development.
— To realize the highlighted above potentials, it is expedient and substantiated to construct the first RI SVBR-75/100 (an industrial prototype) as a part of the NPP unit by the year 2010. In Russian conditions the least cost of it will be if the proposed RI has been installed in the building of the shut down NVNPP second unit with using the existing constructions and some equipment. Carried out estimations have revealed that it will cost ~ $ 100 M.

11. REFERENCES

[1] GROMOV, B.F., TOSHINSKY, G.I., STEPANOV, V.S., et al., 1997, “Use of Lead Bismuth Coolant in Nuclear Reactors and Accelerator-Driven Systems”, *Nuclear Engineering and Design*, **173**, pp. 207-217.

[2] MASAKAZU ICHIMIYA, 2000, “A Conceptual Design Study on Various Types of HLMC Fast Reactor Plant”, *Lead-Bismuth Technology International Meeting* (December 12-14, OEC/JNC/Japan).

[3] ZRODNIKOV, A.V., CHITAYKIN, V.I., TOSHINSKY, G.I., et al., 2001, “NPPs Based on Reactor Modules SVBR-75/100”, *Atomnaya Energiya*, **91**, Issue 6.

[4] GROMOV, B.F., GRIGORIEV, O.G., TOSHINSKY, G.I., DEDOUL, A.V., STEPANOV, V.S., NIKITIN, L.B., 1998, “The Analysis of Operation Experience of Reactor Installation Using Lead-Bismuth Coolant and Accidents Happened”, *Heavy-Liquid Metal Coolant in Nuclear Technology*, (Proceedings, Conference, Obninsk, Russia), Vol. 1, pp. 63-69.

[5] GRIGORIEV, O.G., TOSHINSKY, G.I., LEGUENKO, S.K., 1998, “Bismuth Demand for Commercial Use of RI SVBR-75/100 for Solving Different Tasks”, *Heavy-Liquid*

Metal Coolant in Nuclear Technology, (Proceedings, Conference, Obninsk, Russia), Vol. 2, pp. 556-563.

[6] ROLLAND A. LENGLY, 2000, "Nuclear Industry of the USA in the Transition Period", *Journal of Russian Nuclear Society*, **5-6**, pp. 34-36.

[7] SAVELLY F., 2000, "Nuclear Power Economy in the OESR Countries", *Journal of Russian Nuclear Society*, **5-6**, pp. 36-40.

[8] SIDORENKO, V., CHERNILIN, YU., 2000, "Free Market of Electricity and Possible Consequences", *Journal of Russian Nuclear Society*, **5-6**, pp. 26-33.

[9] PAULSON, C.K., 2002, "Westinghouse AP-1000 Advanced Plant Simplification Results, Measures and Benefits", *ICONE10-22784*, *10th International Conference on Nuclear Engineering*, (Proceedings, Arlington, VA (Washington, D.C.), USA).

[10] POPLAVSKY, V. M., KIRYUSHIN, A.I., SUKNEV, K.L.,et al., 2002, "Prospects of FN Reactors Development", *3rd Scientific Conference of MINATOM in Russia "Nuclear Power. The State and Prospects",* Report, (Proceedings, Moscow, Russia).

[11] RACHKOV, V.,I., 2002, "Nuclear Power Economy", *3rd Scientific Conference of MINATOM in Russia "Nuclear Power. The State and Prospects",* Report, (Proceedings, Moscow, Russia).

[12] WILSON, R., 2000, "The changing need for a breeder reactor", *Nuclear Energy*, **39**, No. 2, pp. 99-106.

[13] ADAMOV, E.O., GANEV, I.H., LOPATKIN, A.V., MURATOV, V.K., ORLOV, V.V., 1999, "Transmutation Fuel Cycle in Large Scale Nuclear Power of Russia", Moscow, GUP NIKIET.

[14] SSC RF IPPE, FGUP EDO "Gidropress", GNIPKII "Atomenergoproekt", 1996, "Determination of the Technical Opportunity and Economical Expediency of Renovating the 2-nd, 3-rd, and 4-th Novovoronezh NPP Units after Exhaustion of their Lifetime with Using Nuclear Steam Supplying Module with Lead-Bismuth Liquid-Metal Cooled Reactor SVBR-75 of 75 MWe", *(TEI)*, Obninsk.

TECHNOLOGY DEMONSTRATION OF PROLIFERATION RESISTANCE FOR AN ADVANCED FUEL CYCLE FACILITY

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Abstract. Argonne National Laboratory and Los Alamos National Laboratory have initiated a joint project on the integration of safeguards in the design of an advanced pyroprocessing nuclear fuel cycle facility. The goal of the project is to develop a demonstrably effective safeguards system for advanced fuel processing technologies via targeted processes and facility modifications and the utilization of modern safeguards techniques. This requires the employment of an integrated design approach that addresses safeguards issues directly during the design stage. The effect of this approach on the safeguardability of the facility will be assessed. In the initial phase of the project, which is currently being completed, the reference pyroprocess facility and preferred safeguards approaches have been identified, along with technology development that will be required in order to further evaluate and demonstrate the safeguards approaches. Future activities are being planned for the demonstration or testing of the key technologies involved. The successful completion of the proposed technology development will demonstrate that a pyroprocessing facility designed with integrated safeguards offers advantages in terms of proliferation resistance and can be satisfactorily placed under international safeguards.

1. INTRODUCTION

Recommendations of the U.S. National Energy Policy Report, written by the National Energy Policy Development (NEPD) Group in May 2001, related to advanced nuclear technologies included: (1) to reexamine the U.S. policies to allow for research, development, and deployment of fuel conditioning methods (such as pyroprocessing) that reduce waste streams and enhance proliferation resistance, and (2) to consider technologies to develop reprocessing and fuel treatments that are cleaner, more efficient, less waste-intensive, and more proliferation-resistant than existing processes.

In both the Advanced Fuel Cycle (AFC) and the Generation IV Nuclear Systems programs of the U.S. Department of Energy (DOE), advanced fuel cycles based on pyrometallurgical processes are under consideration. Argonne National Laboratory (ANL) has been developing a pyrometallurgical process for fast reactor fuels. Such a process is currently being used at the Fuel Conditioning Facility (FCF) at ANL-W in Idaho for the Spent Fuel Treatment Program to process fuel and blanket assemblies from the EBR-II reactor and to generate waste forms suitable for burial in a geologic repository. The treatment (conditioning for disposal) of sodium-bonded spent nuclear fuel at the ANL-W pyroprocessing facility was favourably reviewed [1] in terms of nonproliferation resistance. The FCF facility is also being used in support of the technology development of advanced fuel cycles under the AFC Program.

Pyroprocessing offers specific advantages in terms of proliferation resistance [2]. These advantages include: non-separation of Pu, low decontamination factors (presence of fission products in the product streams), limited transportation requirements when the recycling facility is co-located with the power plants, all processes occurring in one facility (two cells), etc. But, the potential hetereogeneity of some streams used in the process may add a complexity to the implementation of international safeguards. The development of a safeguards approach for such a facility is necessary.

Under the joint sponsorship of the Office of Nonproliferation Policy of the National Nuclear Security Administration (NNSA) and the Office of Nuclear Energy, Science and Technology of DOE, ANL and Los Alamos National Laboratory (LANL) have initiated a project on the integration of safeguards in the design of an advanced pyroprocessing facility. The goal of the project is to develop a demonstrably effective safeguards system for advanced fuel processing technologies via targeted processes and facility modifications and the utilization of modern safeguards techniques. This requires the employment of an integrated design approach that addresses safeguards issues directly during the design stage. The effect of this approach on the safeguardability of the facility will be assessed.

In the initial phase of the project, which is currently being completed, the reference pyroprocess facility and preferred safeguards approaches have been identified, along with technology development that will be required in order to further evaluate and demonstrate the safeguards approaches. Future activities are being planned for the demonstration or testing of the key technologies involved.

2. DESCRIPTION OF THE PROCESS AND REFERENCE FACILITY

The electrometallurgical fuel processing operations used in the project are based on the assumption that the input material is spent fuel from advanced fast reactors. After disassembly of the spent fuel, the individual pins are chopped into small segments and transferred to an electrorefiner, where most of the uranium is extracted and the actinides left in the molten salt. The spent cladding hulls are processed into metallic waste ingots. Salt from the electrorefiner is transferred to a transuranic (TRU) recovery operation that extracts the remaining actinides.The salt, laden with fission products, is consolidated into a ceramic waste form for disposal. The extracted actinides are then processed into new fuel and new elements are fabricated. Three output streams are produced in the facility: new fuel assemblies, metal waste ingots, and a ceramic waste form.

The general picture of the overall process is shown in Figure 1. The Figure indicates the parts of the process that are currently under development or demonstration, such as the TRU extraction, fuel element fabrication, actinide removal from the salt, and a head-end process for the possibility of treating oxide fuel.

The reference fuel is spent sodium-bonded, ternary (U-TRU-Zr) metallic fuel. The throughput of the facility has been designed for a site containing a fuel cycle facility and 4 medium-sized advanced fast-spectrum sodium-cooled reactors, each producing about 800 MW_{th} (~300 MW_e). The reactor type is the metal-fueled sodium fast reactor option under consideration in the Generation IV program. Each reactor unit produces approximately 3 MTHM (Metric Tonne of Heavy Metal) per year, for a fuel cycle facility throughput of 12 MTHM per year.

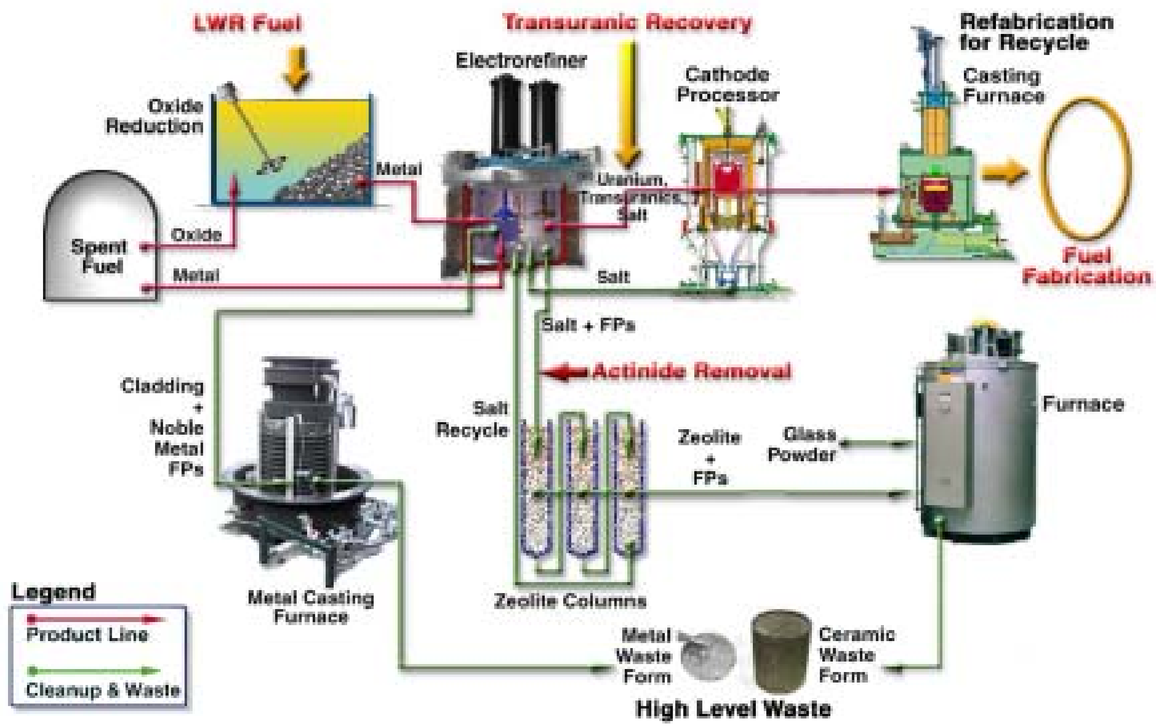

FIG. 1. Flow diagram of the general pyrometallurgical process.

Consistent with the initial missions of Generation IV fast spectrum reactors in the area of actinide management, the reactor cores are designed for net actinide burning (a conversion ratio of ~0.72 was assumed although lower values are feasible). In this operating mode the reactors are not fissile self-sufficient and require a source of makeup fissile material. Three recycle options were initially considered for the facility, differing in the source of the makeup stream. After evaluation, an option assuming that feed material comes from a recycle of twice-through light water reactor (LWR) fuel in the form of mixed oxide (MOX) was selected. The original LWR fuel is assumed to have a burnup of 50 GWd/MT. This is one of the recycle scenarios that has been considered in the systems studies in the AFC Initiative. The makeup has a large concentration of high actinides, Cm in particular, and it was chosen because it presents the greatest challenge for neutron counting with a Pu non-destructuve assay (NDA) method because of the relatively high Cm content.

Spent fuel pins are chopped and the pieces are placed into electrorefiner anode baskets for processing. From this point on all the process needs to be conducted in an inert atmosphere, since the sodium bond (and other pyrophoric materials) are exposed.

At the center of the pyroprocess is the electrorefiner [3], where the uranium in the spent fuel is electrochemically separated from the other spent fuel constituents. This is accomplished by passing a current from the baskets containing the chopped spent fuel to a solid cathode surface. A molten salt serves as the electrolyte for the electrorefiner. The uranium, the TRUs, the active metal fission products, the bond sodium, and a small amount of zirconium dissolve from the anode basket into the molten salt, while only uranium and a small amount of contaminants collect at the cathode surface. The noble metal fission products and the cladding remain behind in the anode baskets. Species not collected at the cathode surface remain dissolved in the molten salt in the form of chloride salts.

The Uranium Product Processing Unit separates adhering salt from the uranium product by vacuum distillation and consolidates the uranium product into a solid metal ingot. The uranium product is processed in batches. This uranium product is transferred to the injection casting furnace for further processing into new fuel, when properly mixed with TRU feed.

Pu and minor actinides in the electrorefiner remain in the salt, along with some residual uranium.The salt is transferred to the TRU extraction step, which removes essentially the totality of the actinides, with only fission products and traces of actinides remaining in the salt. The U/TRU product is processed as the U product, removing the remainng adhering salt by vacuum distillation.

The metal waste is processed in a manner very similar to the Uranium Product Processing Unit, with vacuum distillation. The metal waste form is sampled, put into containers, sealed, and sent away for disposal. In the Ceramic Waste Processing Unit, glass frit, ground zeolite, and waste salt are combined together in a mixer operating at 500 ºC to form a well-combined mixture of salt, zeolite, and ground glass frit. The mixture is then poured into waste containers and is heated to above 800 ºC for an extended period of time that allows the material to consolidate and melt into a ceramic waste material that locks in the salt and salt contaminants. The waste containers are then sealed and sent away for storage and disposal.

New fuel pins are fabricated by injection casting. In the injection casting step, fuel material (i.e., U and U/TRU products plus fissile makeup material) is loaded into a crucible, which is then placed into an injection casting furnace. The pin castings are then removed from the molds, inspected for correct weights and dimensions, and placed within an element clad, which is then bonded with sodium, welded, leak tested, and inspected. Finally, in the assembly fabrication step, a specified number of fuel elements are assembled into a fuel assembly.

A reference facility layout has been developed as a result of an iterative process. An initial layout was proposed. Upon identifying the safeguards options for the facility, modifications were made to facilitate their implementation, but accounting for facility operation and maintenance constraints. The reference facility layout is shown in Figure 2.

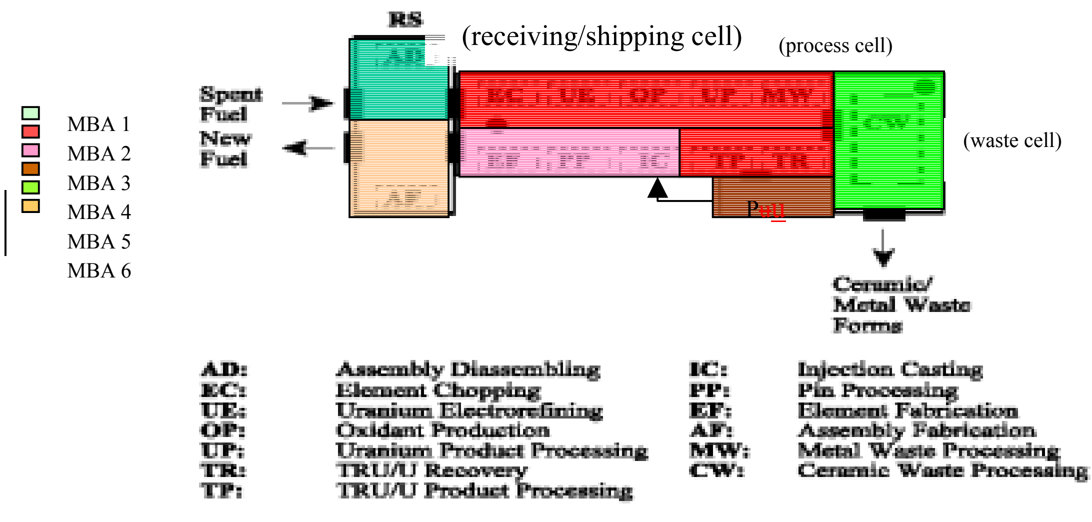

FIG. 2. Reference facility layout including Material Balance Areas (MBA).

3. SAFEGUARDS APPROACHES

In order to develop international safeguards approaches for the reference pyroprocessing facility, a thorough review of the experience in appying International Atomic Energy Agency (IAEA) safeguards to reprocessing facilities was conducted [4]. The specific IAEA safeguards application is highly dependent on the process and the facility design. Although the overall goals and the general principles of the IAEA safeguards are applicable to all facilities of the same type, the specifics for a particular facility are negotiated by the IAEA on a case by case basis.

The overall objective of the IAEA is to independently verify the declarations made by the State on the uses of nuclear material. As a general rule the IAEA safeguards system consists of two major activities:

(i) nuclear material accountancy (NMA) – the basic tool for safeguarding nuclear material. It establishes the amount of material in the facility, documents changes in inventories, and quantifies material unaccounted for. Measurements are a key part of the system, and material balance areas (MBA) and key measurement points (KMP) are defined to facilitate the measurement of inventories and flows.

(ii) Containment and surveillance (C/S) – methods used to complement material accountancy. It includes material enclosures and penetration monitoring.

At the only existing (aqueous) reprocessing facility under safeguards, the IAEA uses a combination of NDA, DA, containment and surveillance and process monitoring. There are technical challenges (measurement uncertainties, for example) in implementing IAEA safeguards in large reprocessing facilities. Pyroprocessing facilities, for which there is no experience with application of IAEA safeguards, will require innovative approaches, as some of the standard methods used in aqueous facilities may not be applicable. Development of equipment and procedures specifically for pyroprocessing may be required.

Because pyroprocessing is a relatively new technology, innovative safeguards approaches have been studied. In this exercise, four alternative approaches have been identified [5] for further assessment. The approaches are described below.

3.1. Proposed MBAs

In the initial iteration, the facility layout included four material balance areas. After consideration of possible safeguards approaches, a variation of the layout consisting of six MBAs has been selected to facilitate the implementation of safeguards (see Figure 2):

— MBA 1: RS – Receiving/shipping, spent fuel only (to simplify surveillance)
— MBA 2: PC – process cell (physically separate from all other facility activities)
— MBA 3: Fresh Fuel hall
— MBA 4: Pu Store (dual C/S)
— MBA 5: WC- Waste cell (it could optionally be merged with PC)
— MBA 6: RS – Receiving/shipping, Fresh Fuel (to simplify surveillance)

The pyroprocessing facility design integrates into a single facility a fresh and spent fuel handling/assembly facility, a reprocessing facility, a fuel fabrication facility, a separated Pu store, and a waste processing facility. (Four reactors would also be co-located on the site.) While the six proposed MBAs are two more than the initial four envisioned in the project, they are less than the number of MBAs typically associated with the individual stand-alone facilities. From a safeguards, material control and accountability (MC&A) and C/S perspective, the separation of the receiving/shipping and process cells into multiple MBAs according to the material types in each area is the preferred layout.

However, the requirements for physical separation of the MBAs leads to potential design and operability implications and the design of the facility may need to be adapted accordingly. The requirement for physically dividing the receiving/shipping cell or process cell to support the MBA structure, may add extra transfer ports, and may require additional remote handling equipment. Extra transfer ports to the hot repair area would be needed so that equipment from all of the MBAs could be transferred out of the cell for maintenance and repair. A third potential design change imposed by multiple MBAs in the two cells is in the cell cooling and ventilation systems. As part of the entire design approach, trade off studies will be conducted to optimize the implementation of the safeguards with the operational and maintenance requirements.

3.2. Safeguards options

The focus of the project is on MBA 2, the process cell. While other MBAs such as the fresh fuel hall (MBA 3) present unique issues, it is believed that the most pressing safeguards elements will be faced in this MBA.

As a means of structuring the design process, and in order to develop a range of possible NMA and C/S combinations, each point in a detailed facility flow diagram was examined as a possible KMP for an initial Pu accountancy measurement. Such an approach produced a set of alternatives varying in wide degree in terms of their reliance on containment and surveillance. Those approaches that defer an initial Pu assay until later in the process, must rely more heavily on C/S measures. It must be noted that all of the options presented here are different from current IAEA practices at aqueous facilities.

After an initial screening, four safeguards options were identified for further consideration in the project. Of the five possible KMPs for initial Pu accountancy, two were eliminated as being clearly inferior to other approaches. In addition, an alternative that included a modification of the proposed process to accommodate a spent fuel homogenization step was added as it could provide significant safeguards measurement advantages.

3.2.1. Safeguards Option 1: Neutron Balance - Cm accounting

This approach involves the following elements:

— Total neutron measurement on each pin entering MBA 2 for continuity of knowledge (COK) rather than Pu assay purposes;
— Total neutron measurement on electrorefiner to determine Pu inventory via a Cm ratio technique;
— Total neutron measurement of waste streams and U product leaving MBA 2;
— NDA or destructive analyses (DA) (assuming homogeneity) on U/TRU product to obtain Pu assay for transfer to MBA 3 and a Pu/Cm ratio
— Process monitoring of the electrorefiner (ER) and the oxidant production unit (OP) to provide confidence that Pu and Cm remain commingled
— Integrated video and neutron monitoring of material transfer pathways and MBA penetrations

The safeguards concept would be to use neutron measurements to maintain COK for the Pu/Cm mixture in the process until a quantitative Pu accounting measurement is made on the U/TRU product stream. The neutron producing isotopes input to the process cell, less those isotopes lost in waste streams and resident in the electrorefiner, should appear in the U/TRU product stream.

In effect, Cm is accounted for as a temporary surrogate for Pu. Therefore in this approach process monitoring will be critical to ensure that Pu is not separated from Cm at any point in operations. As individual pins are fed into the chopper, a total neutron count would be taken to establish the neutron emission rate for each pin. Integrated video and neutron monitoring around the ER to monitor incoming and exiting streams would ensure that KMPs are not being bypassed. Process monitoring obviously plays a critical role in this approach. Process parameters would be monitored, but the Pu/Cm inseparability argument would have to be proven.

This option is the most straightforward in terms of measurements and would have the least impact on operations. From a facility design perspective, this option is the easiest to implement.

3.2.2. Safeguards Option 2: Electrorefiner Assay

This approach involves the following elements:

(i) Pu input accountability data are obtained utilizing the following "batch tracking" technique around the electrorefiner (ER):
  - Assay the Pu content of all U cathodes removed from the ER using a Cm ratio technique similar to that used at the Tokai Reprocessing Plant or DA of U product after processing each day
  - Assay the Pu content of all metal waste baskets removed from the ER during a day using a Cm ratio technique
  - Assay the Pu content and the Pu/Cm ratio in the electrorefiner salt prior to daily salt removal using a DA sample (assuming a homogeneous mixture is obtained)
  - Weigh the ER salt removed daily and determine Pu content based on the DA sample composition
  - Assay the Pu content of the recharge salt using a Cm ratio technique
  - Assay the Pu content of the recovered salts from the metal waste and U product processing units using a Cm ratio technique

(ii) The remaining data necessary for closing a material balance on MBA 2 is obtained using the following measures:
  - DA (assuming homogeneity) on U/TRU product for Pu content
  - Assay the Pu content of the metal waste ingots and waste salt containers using a Cm ratio technique

(iii) Process monitoring on ER and OP to provide confidence that Pu and Cm remain commingled

(iv) Integrated video and neutron monitoring of material transfer pathways and MBA penetrations

This approach involves essentially closing a material balance on the electrorefiner each day, coincident with Pu loaded salt removal from the ER, in order to derive Pu input accountability data. Combined with Pu data from output streams and the ER inventory, a material balance can then be closed for MBA 2 on a daily basis if necessary.

This approach will rely on a high level of surveillance to ensure that KMPs are not bypassed (e.g., salt from ER to electrolysis unit, etc.). Combined video and neutron radiation monitoring could be useful in this regard in monitoring material transfers.

Each batch analysis step (e.g., cathode counting, ER salt counting) adds a delay between processing steps that could significantly reduce the processing capacity of the facility. This impact must be evaluated. The ultimate accuracy in the measurement of the Pu content in the ER is unknown and will likely dominate the overall material balance uncertainty. Salt homogeneity is an important issue in this context.

3.2.3. Safeguards Option 3: Homogenized Input

This approach involves the following elements:

— Add a homogenization step (e.g., fine shredding and mixing) after the element chopping step
— Produce a homogeneous sample for DA to obtain Pu composition for Pu input accountability and also obtain Pu/Cm ratio for potential use in downstream assays
— Combine Pu concentration data and volume measurements of homogenized material to obtain Pu input accountability data
— Total neutron measurement on electrorefiner to determine Pu inventory via a Cm ratio technique
— DA (assuming homogeneity) on U/TRU product for Pu content and a Pu/Cm ratio for use in downstream assays
— Assay the Pu content of the metal waste ingots and waste salt containers using a Cm ratio technique
— Process monitoring on ER and OP to provide confidence that Pu and Cm remain commingled
— Integrated video and neutron monitoring of material transfer pathways and MBA penetrations

This approach attempts to facilitate the use of DA in determining Pu input accountability. Such an approach would mirror more closely the IAEA's current approach for aqueous facilities. This option would require that the input material (spent fuel pins) be homogenized by a fine shredding and mixing process or other means. The resulting material could then be sampled for DA. Combined with a total mass measurement, the Pu in that input batch could be determined. As in other options, the product and waste streams would be analyzed for Pu content perhaps using a Cm ratio technique.

Variations in Pu/Cm ratios in the input spent fuel may require the use of batch tracking similar to that employed in Option 2 to determine ER inventory. (If the Pu/Cm ratio in the ER differs appreciably from that measured in the homogenized material it cannot be used to determine Pu inventory in the ER.) The Pu/Cm ratio from a DA sample from the ER could be used for downstream assays.

As with the other options, this approach will rely on surveillance to ensure that KMPs are not bypassed. This option would require a modification of the pyroprocessing technique as developed to this point. It would also require an assessment of the feasibility of obtaining a sufficiently homogeneous mixture after fine shredding and mixing of the fuel pins.

3.2.4. *Safeguards Option 4: Assay of Pu in Spent Fuel via Pu/Cm Ratio and Destructive Analyses*

This approach involves the following elements:

— Total neutron axial profile on each pin entering MBA 2 used to establish a Cm profile
— DA on a certain number, to be determined, of select pieces of rods to determine Pu/Cm ratio; used to produce Pu input accountability data on a pin by pin basis
— Total neutron measurement on electrorefiner to determine Pu inventory via a Cm ratio technique
— Total neutron measurement of waste streams and U product leaving MBA 2
— NDA or DA on U/TRU product (assuming homogeneity) to confirm Pu/Cm ratio and provide Pu assay for transfer to MBA 3; this ratio could also be combined with the total neutron measurement on the ER to determine ER inventory
— Process monitoring on ER and OP to provide confidence that Pu and Cm remain commingled
— Integrated video and neutron monitoring of material transfer pathways and MBA penetrations

This option is a variation of Option 1. The difference is the basis used for Pu input accountability. Rather than use the Pu/Cm ratio obtained at the end of separations to produce Pu input accountability data, a Pu/Cm ratio is obtained for each spent fuel pin input to the MBA. The neutron profile taken on each pin should allow an extrapolation of DA results along the axis of the pin with minimal reliance on reactor modeling. However, this technique will have to be demonstrated as a means for producing accurate Pu input accountability data. An advantage of this method over Option 1 is that the Pu/Cm ratio is measured at two points (i.e., pin segments and U/TRU product) in the process. This enables a more targeted use of each ratio.

This option is relatively easy to implement in the facility design. A critical parameter to be determined is the throughput for performing the axial scan of each pin. Depending on the counting statistics required, multiple detectors or multiple counting stations might be required.

3.2.5. *Process monitoring*

Process monitoring is focused on providing independent confirmation that processing units are being operated as declared. Key process parameters of the electrorefiner and the oxidant production units should be monitored. Process parameters are being monitored and controlled for operational purposes in the reference pryroprocessing facility. This extensive process monitoring can support the safeguards approach if independent verification of the measurements can be ensured, either with the use of independent monitors or dual-use sensors. The most relevant process parameters in the two key steps are described below.

3.2.5.1. Electrorefiner

In the electrorefiner, three process variables, namely cell voltage, cell current, and species concentrations, can provide direct information supporting safeguards objectives. Two additional process variables, namely salt level and density, can be used to compute confirmatory/calibrating safeguards data. The TRU extraction step will also require process monitoring.

3.2.5.2. Oxidant Production Unit

The oxidant production unit involves chemical reactions. To ensure that the facility is actually producing oxidant material for proper operation of the electrorefining step, the flow of chlorine gas to this equipment should be monitored. In addition, online monitoring of U concentration in the salt via voltammetry techniques can provide an indication that the chemical reaction is occurring during oxidant production.

3.2.5.3. Mass Tracking System

The Mass Tracking System (MTG) has been designed to track the movement and location of materials inside the Fuel Conditioning Facility (FCF) and the Hot Fuels Examination Facility (HFEF) at ANL-W in Idaho. The system aids operations personnel in process control, materials accountability, and compliance with both facility operating limits and criticality safety limits. Tracking is done by container or item, material form, and material type.

An advanced fuel recycle facility based on pyroprocessing would use an extension of the MTG system. The enhanced MTG could include automation of material movements to both track materials and minimize the potential for diversion within the facility, including an automatic container identification system. Remote handling equipment can be controlled by the MTG to perform unattended operations that follow strict movement paths. Non-destructive assay stations along the movement paths can be incorporated into the system.

For process monitoring, the MTG can interface with the facility data acquisition system (DAS) to monitor important process parameters that have safeguards implications, such as the ratio of Pu to U in the ER, integrated current for a given ER run, etc. The modular nature of the MTG makes it easy to accommodate additional functions without restructuring the entire system.

4. CURRENT STATUS AND FOLLOW UP WORK

Four safeguards options have been identified. The options are currently being assessed from two points of view, namely (1) safeguards, for which an acquisition path analysis will be performed, and (2) impact on the facility design and operations.

Because all the proposed safeguards options rely on innovative approaches, the technologies to be used need to be completed and/or demonstrated. Specific technology development needs to support the proposed safeguards approaches have been identified. The list of technology needs includes:

(i) Fuel pin Pu/Cm ratio measurement and sampling issues. This may include sample fabrication and testing.
(ii) Development of approaches for obtaining representative samples of the fuel inserted in the facility, such as shredding, blending, sampling strategy, and equipment needs.
(iii) In situ assay of the electrorefiner.
(iv) Process monitoring (voltammetry and others).
(v) Waste stream NDA, as part of NRTA general approach.

A Pu/Cm ratio technique is utilized to a greater or lesser extent in each of the safeguards options. For streams where Pu content is expected to be low, its utility seems clear. When it is to be used to provide accountability data on significant amounts of Pu its accuracy must be verified. If the Pu/Cm ratio is to be utilized extensively, it must be demonstrated that the process is not producing variations in this ratio. This places a high importance on process monitoring and confirmatory measurements.

As development or demonstration of the necessary technologies progresses, a preferred safeguards approach selection will be possible. In the selection process, trade offs between the safeguards approach and the impacts on the facility will need to be assessed.

The existence of an operational facility for pyroprocessing at ANL-W provides a unique opportunity to support the safeguards technologies identified for development and demonstration. Moreover, the facility has been operating for a significant length of time and has been used to support the development of pyroprocessing. This has resulted in the accumulation of data that can be very useful in support of the safeguards technology development.

5. CONCLUSIONS

Pyroprocessing offers potentially attractive features in terms of proliferation resistance but some specific characteristics, such as the potential hetereogeneity of some streams used in the process, may require innovative approaches to the implementation of international safeguards. The current project has defined a reference pyroprocess facility and has identified four safeguards approaches for the facility. The current reference layout for the facility is already the result of an iteration between process facility design and the proposed safeguards approaches.

The safeguards options rely on innovative technologies that will need development or demonstration in representative conditions. Development and demonstration of these technology needs is being proposed as a follow up to the current phase of the project. After the innovative technologies have been developed sufficiently, it will be possible to select the preferred safeguards approach and update the reference facility layout for consistency with the selected option.

In establishing the resulting safeguards and facility design, facility characteristics that may enhance the implementation of the safeguards will be used to the extent possible. Because all the process occurs in a single cell, under inert atmosphere, the number of penetrations is limited. Furthermore, process parameters in key steps can be monitored with dual use or dedicated sensors. An advanced mass tracking system will also be part of an advanced pyroprocessing facility. Given the importance and possible advantages of the reference facility in terms of surveillance and material tracking, these techniques should be leveraged to the greatest extent possible. While radiation monitoring of paths and penetrations is referenced above, the possibility of area coverage should also be explored. It is understood, naturally, that ultimate acceptance by the IAEA can only be established by negotiations for a specific installation.

The successful completion of the proposed technology development will demonstrate that a pyroprocessing facility designed with integrated safeguards offers advantages in terms of proliferation resistance and can be satisfactorily placed under international safeguards.

6. REFERENCES

[1] U.S. DEPARTMENT OF ENERGY, Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel, Office of Arms Control and Nonproliferaiton, Washington, D.C., July 1999.
[2] HANNUM, W.H., WADE, D.C., MCFARLANE, H.F., HILL, R.N., Nonproliferation and Safeguards Aspects of the IFR, Progress in Nuclear Energy, **31** (1997) 203.
[3] LAIDLER, J.J., BATTLES, J.E., MILLER, W.E, ACKERMAN, J.P., CARLS, E.L., Development of Pyroprocessing Technology, Progress in Nuclear Energy, **31** (1997) 131.
[4] LONGMIRE, V., LI, T.K., THOMAS, K.E., IAEA Safeguards Approach for Reprocessing Facilities, Los Alamos National Laboratory, LA-UR-02-5276 (2002).
[5] BUDLONG-SYLVESTER, et al., International Safeguards for Pyroprocessing: Options for Evaluation, Los Alamos National Laboratory, LA-UR-03-0986 (2003).

IMPORTANCE OF INHERENT SAFETY FEATURES AND PASSIVE PREVENTION MEASURES IN INNOVATIVE DESIGNS

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Abstract. Both inherent safety features and passive safety systems can help to achieve the safety requirements of INPRO and GenIV and may improve the public acceptance of nuclear power. Passive systems for heavy and light water reactors can help to reduce the core damage frequency, and thus, also the investment risk. Moreover, they can help to reduce the capital cost of new reactor systems through simplifications in the design. However, passive components / systems require extensive experimental and analytical investigations to ensure their reliability. Additionally, in-service inspections are needed. Ideally, the passive approach is completely independent of the active one and, thus, meets the diversity and redundancy requirements. A combination of both an active and a comprehensive passive approach appears to be optimal. As future systems are supposed to meet the sustainability criteria i.e. breeders or molten salt reactors, more inherent safety features are needed. These include e.g. a low-pressure coolant, low power densities, small control reactivities and negative feedbacks that lead to a self-shutdown in critical accident situations. Furthermore, comprehensive passive safety systems are of course also desirable for these advanced systems. These improved safety approaches appear necessary in light of a potentially rapid increase of nuclear power in the next half of this century.

1. INTRODUCTION

Many passive components and approaches have been included in the designs of new water or heavy water reactors. This concerns on the one hand measures for detecting failures and controlling and terminating accident sequences e.g. by emergency cooling. On the other hand, the mitigation of severe accident phenomena such as core melts is also included in modern reactor designs. For highly advanced designs that can breed, more inherent safety features are proposed in addition to passive measures.

The IAEA has issued a document "Safety Related Terms for Advanced Nuclear Plants"[1] and IAEA staff has given an overview on reactor designs with passive safety features [2]. The following definitions are given in these papers:

— Passive Component: A component which does not need any external input to operate.
— Passive System: Either a system which is composed entirely of passive components and structures or a system which uses active systems in a very limited way to initiate subsequent passive operation.
— Inherent safety characteristic: Safety achieved by the elimination of a specified hazard by means of the choice of material or design concept.

Large evolutionary LWR designs such as EPR and ABWR incorporate proven active systems for accomplishing safety functions. Diverse and redundant components of proven high reliability are used [2]. However, the EPR design also contains passive autocatalytic recombiners for preventing a hydrogen detonation during a core degradation phase as well as an external core catcher for passively cooling a core melt. These passive features are relevant for meeting the INPRO user requirements Level 4: Prevention of Major Radioactive Release and Level 5: Prevention of Containment Failure and Mitigation of Radiological Consequences [3].

For designs in the 1300-1500 MWe range, consideration is given to incorporation of passive systems. For example the Mitsubishi 1400 MWe APWR design contains accumulators for injecting cooling water until the core is reflooded in a Loss-of-Coolant accident (LOCA). For their next generation PWR in Japan with 1500 MWe, the inclusion of hybrid systems combining both active and passive systems is being investigated [2]. The design of the Next Generation PWR in Korea, the APR1400, has a passive system to inject water into the reactor coolant system and passive autocatalytic recombiners [4].

Ref. [2] gives an overview of the extensive passive safety features of reactor designs between 900 and 1300 MWe. This includes the EPP-1000 (Westinghouse USA, Genesi, Italy), VVER-1000, now featuring an ex-vessel core catcher, the ESBWR (GE, US), SWR-1000 (Framatome-ANP) and CANDU-9 (AECL). Smaller passive designs are also discussed in Ref. [2]. These are the AC-600 (Advanced Chinese PWR), the Westinghouse design AP-600, the Mitsubishi MS-600, the Hitachi Simplified BWR HSBWR, the CANDU-6, the Indian Advanced Heavy Water Reactor AHWR, the PIUS concept of the former ABB Atom and the somewhat related ISIS design of Ansaldo, Italy. These passive designs include usually large tanks of water inside the containment for safety injection. Long-term injection water is provided by gravity from the large IRWST (In – containment Refuelling Water Storage Tank). The latter is also used as a heat sink for emergency decay heat removal. Once boiling in the IRWST starts, the passive containment cooling becomes important. The PIUS and ISIS designs are different. In these designs the core and regular circuit are surrounded by heavily borated water. The latter will penetrate the regular circuit in case of a LOCA, overheating and boiling during a station black-out or in case of transients e.g. caused by feedwater problems. An interesting new passive feature is the inclusion of the steam generators inside the vessel of a PWR. Any water lost from a leak in the primary circuit will therefore remain in the vessel. Such an integral design is IRIS (Integral Reactor Inherently Save) [26].

None of the advanced designs with extensive passive safety features have been built or ordered yet. This may be related to the fact that comprehensive passive systems were not available earlier (some manual activation of passive systems was still needed and larger systems have passive devices only for mitigating severe accidents). But there were also questions about the inspectability of passive components, whose failure modes can be subtle and may be missed (i.e. a heat transfer may be impeded by a blanket of non-condensable gases); their operation may be “safe” but may stress the plant components more heavily than an active system. Also they tend to have large volumes that may be costly [3]. However, due to large experimental programs on passive safety devices and due to new designs that have shown that passive safety can reduce cost, some of these arguments are no longer valid. Also the accessibility of passive systems for in-service inspection has been improved.

2. EXAMPLE OF AN LWR DESIGN WITH A COMPREHENSIVE AND EXPERIMENTALLY TESTED PASSIVE SYSTEM

The SWR-1000 evolutionary BWR design appears to be the first reactor available on the market equipped with a comprehensive passive safety system in addition to the active one. This means that a reactor operator is not urgently needed to prevent core damage in case of a severe accident initiator. Due to the large grace period of 72 hours, the relevance of the failure of appropriate operator intervention is negligible. The passive devices included are [5], [6]:

— Passive pressure pulse transmitters that sense a decrease of the water level and actuate reactor scram, reactor depressurisation and isolate the containment by closing the steam isolation valves.
— Passive flow limiters in the feed water lines
— Hydraulic scram system for rapid control rod insertion as well as a fast-acting boron injecting system for reactor scram
— Additional passive spring-loaded pilot safety relief valve for reactor pressure relief and depressurisation
— Emergency condensers for heat removal from the reactor pressure vessel (RPV)
— Containment cooling condensers for containment heat removal. The heat is transferred to a large dryer-separator storage pool above the RPV
— Flooding lines for passive core flooding in the event of a LOCA. The water comes from the large core flooding pools.
— Drywell flooding system for cooling the RPV exterior for in-vessel melt retention in the event of a core melt accident – the flooding is driven by gravity, but the valves are actively actuated since an erroneous RPV flooding could lead to thermal stress problems.
— Nitrogen-inerted containment atmosphere to preclude hydrogen combustion in the case of a severe accident. This feature is also common in other BWR designs.
— It should also be mentioned here that BWRs have a strong negative feedback when the coolant temperature increases since more boiling will occur. This is relevant in Non-LOCA transients.

The complete SWR-1000 safety approach is described in Figure 1.

The advantages of this system are according to [7]: 1.) Simplification of systems engineering, 2.) Reduction of dependence on external power systems, 3.) Significant reduction in effects of common-cause faults, 4.) Low susceptibility to human error, 5.) Lower cost and resource requirements for inspection and maintenance, 6.) Reduced probability of core melt.

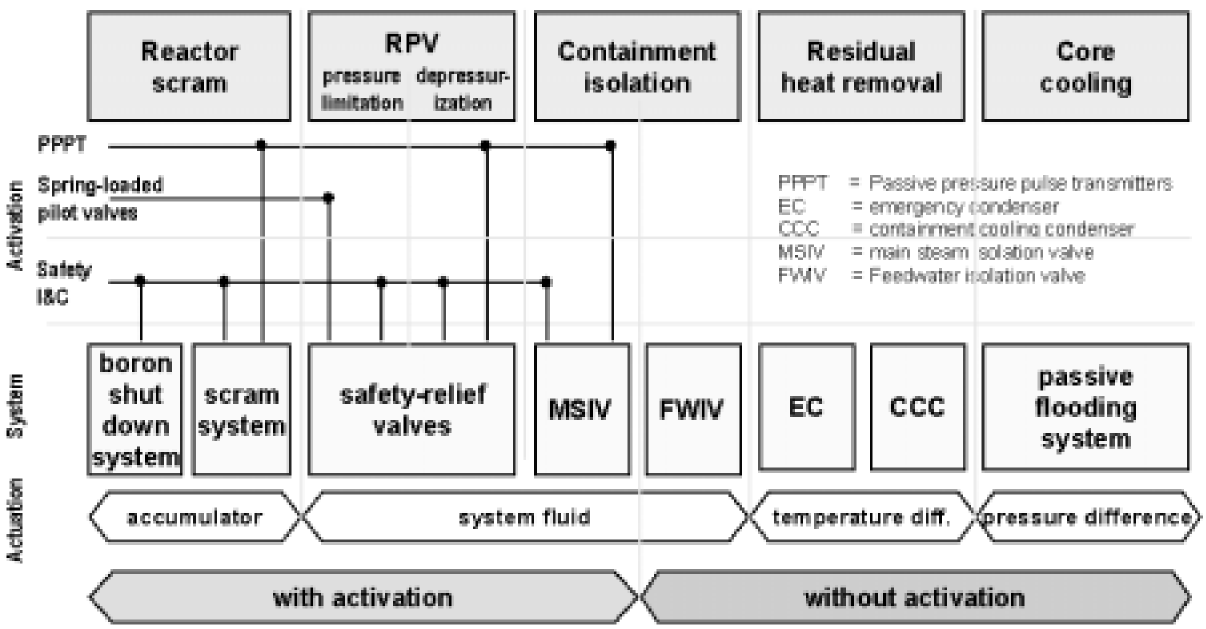

FIG. 1. The passive safety equipment for the different safety functions – from [6].

2.1. Experimental test program of passive systems

A good understanding of a passive system requires extensive testing of its different components. Extensive theoretical and experimental investigations have been performed for SWR-1000 on the passive pressure pulse transmitter, the emergency condenser, the passive outflow reducer, the passive core flooding system, the containment-cooling condenser and the in-vessel retention of core melt by cooling the RPV exterior.

It is notable that in the testing of the passive pulse transmitter it was found that the response time could be reduced considerably to 3 s at 70 bar and 10 s at 10 bar. The passive outflow reducer is actually a safety feature of the passive emergency condenser since it limits the outflow from this condenser in case of a guillotine break in one of the condensate return lines. In testing the containment-cooling condenser it was confirmed that the nitrogen that first surrounds this condenser gets swept by the steam into the wet well via a venting line. It was also found that in the case of core degradation the generated hydrogen formed stratified layers that could not be simulated satisfactorily by all computer codes. Moreover it was found that fins at the outside of heat exchanger pipes are not necessary for condensing steam and that neglecting them made the component, the experiments and the analysis simpler.

2.2. Probabilistic ris\k analysis

In [5] the integral frequency of core hazard states resulting from plant-internal events occurring during power operation is reported to be around 6×10-8 for SWR-1000.

The credibility of such low numbers is often questioned as they are reaching the general limits of historical experience of events. For example, earthquakes above a certain magnitude can be bounded based on historical evidence to less than 10-3 or perhaps 10-4 /year.

When operators are properly trained and motivated, such as airline pilots, there is evidence that estimates of trained human operation inattention to a control task in the area of 10-9 per flight are realistic (due to large amounts of statistical data available). This would indicate that human operators can perform quite well, but not perfectly under these conditions; however, it is doubtful that NPP operators would have a better error rate in emergency situations regardless of how good or automated the design.

Therefore, operator actions which have direct core damage impact (for example, failure to go to recirculation) should have probabilities limited by the best observed human performance, such as those documented for commercial airline pilots. Thus, predictions of core damage, which require human error frequencies of the order of 10-7, are clearly incredible because the historical data set required for establishing such human performance is non-existent.

However, for plant designs in which the human performance on the core damage is negligible (e.g. due to the large grace period in the case of the SWR-1000), the very low core damage frequency value mentioned (about 6×10^{-8}) is in principle credible.

A more specific argument supporting the credibility of this low core damage frequency is related to the input assumptions used in the underlying PSA. For reactors with passive system functions, the failure of the physical process itself usually is the major contributor to the failure of the whole system together with some activating components. The quantitative modeling of the reliability of a physical phenomenon, e.g. a passive core flooding device is usually performed by deriving estimates from thermal hydraulic modeling. Therefore, an essential element for validating such PSA studies is the assessment of the uncertainties of the deterministic calculations. They also play a decisive role in the determination of the success criteria in a PSA. To the knowledge of the authors, this has been done for the SWR-1000 right from the beginning of the design-specific PSA evaluations.

2.3. Economics of passive safety

Regarding the economics of the SWR-1000 it is stated that the simplified safety concept combining active and passive systems together with the simplified system for normal plant operations enables a reduction in capital cost of 30 % compared to existing light water reactor designs. Due to the simplifications, maintenance costs are also lower since fewer components have to be inspected, maintained and repaired, and thus, fewer man-hours are needed for these activities [5].

3. INHERENT SAFETY APPROACHES

For more advanced designs that can meet the sustainability criterion by breeding and burning their own waste, fast systems or molten salt reactors are needed. The belief that they need to be introduced only in several decades is risky. A significant increase in nuclear power, that is needed to curb the generation of greenhouse gases and make countries less dependent on oil and gas, will probably lead to an exhaustion of the known and speculative reserves of fissile uranium by the middle of this century [9]. Moreover, fast systems, MSRs and in particular accelerator-driven systems (ADSs) together with an efficient reprocessing can strongly reduce the long-lived higher Pu-istopes and minor actinides that are an important part of the nuclear waste problem.

In particular fast reactors contain multiples of the critical fuel mass. This may lead to super-critical conditions in a core-melt scenario. Furthermore, they tend to have a high power density, which is due to the compact design required to reduce the amount of fissile material. Because of these intrinsic safety concerns, other inherent safety aspects are needed. Earlier in this paper the definition for an inherent safety characteristic was given: “Safety achieved by the elimination of a specified hazard by means of the choice of material or design concept”. The water coolant of LWRs actually has some intrinsic safety aspects such as a negative voiding coefficient, a high heat capacity and a good natural circulation capability. However, the necessary high coolant pressure is the main reason that LOCAs are significant accident initiators for LWRs. Low-pressure coolants such as sodium, lead or lead/bismuth and molten salt don’t have this problem. But sodium coolant, despite its good heat conductivity and natural circulation capability has a positive void reactivity together with a not too high boiling point (892 °C) and this has always been a concern. Additionally there is a strong interaction with both water and air and there also some remaining concerns about corrosion.

3.1. Inherent safety aspects and passive prevention approaches in heavy metal-cooled systems

For lead based coolants the voiding is a minor concern because the boiling point is so high (for pure lead 1737 °C). Even in an overpower condition due to the withdrawal of all control rods at zero reactor power and no subsequent reactor scram, the outlet temperature in the SVBR75/100 reactor [10] stays at about 860°C. For larger heavy metal cooled systems such long overpower conditions may rather lead to vessel creep than core voiding. Another advantage of lead based coolants is that they are chemically inert against water and air. Nevertheless, a heat exchanger with superheated steam in the tubes and located in a pool type vessel requires a special design to accommodate possible steam tube leaks [10, 15]. The neutronic characteristics of lead-based alloys lead to a lower core leakage, and thus, a lower enrichment and a considerably lower burn-up swing than sodium. The latter aspect allows a total control rod worth of less than the delayed neutron fraction (1$) during a few years of burn-up. This is significant for safety since it makes large reactivity insertions from control rods impossible. But lead based alloys have also a lower Doppler effect than sodium (-0.098 vs.-0.173 cents/ °C for a 300 MWe core) because of its harder spectrum [11]. Nevertheless, the automatic self-shutdown in unprotected Loss-of-Flow (LOF) or Loss-of-Heat-Sink (LOHS) accidents due to negative Doppler and axial fuel expansion feedbacks is assured [10], [11], [12]. This self-shutdown had been demonstrated earlier in the EBR-II for LOF and LOHS accidents without scram [13]. Furthermore, the low leakage of lead cores allows wide coolant channels so that the coolant can occupy more than 50 % of the core volume. This in turn decreases the core power density. Moreover, lead is also a very good reflector, and thus, the surrounding structures will be rather well shielded [11, 14]

A passive feature of a heavy metal cooled-reactor is a Reactor Vessel Air Cooling System (RVACS), which had been developed earlier for Na-cooled reactors. It is a continuously operating system that is needed for emergency decay heat removal e.g. in a protected station blackout accident in which the diesel generators couldn't get started for a forced convection cooling. Figure 2 shows that an 800 MWt modular Pb/Bi cooled (Accelerator Driven System) ADS with a vessel height of 17 m and a diameter of 6m could be well cooled with a PRISM-type RVACS [15].

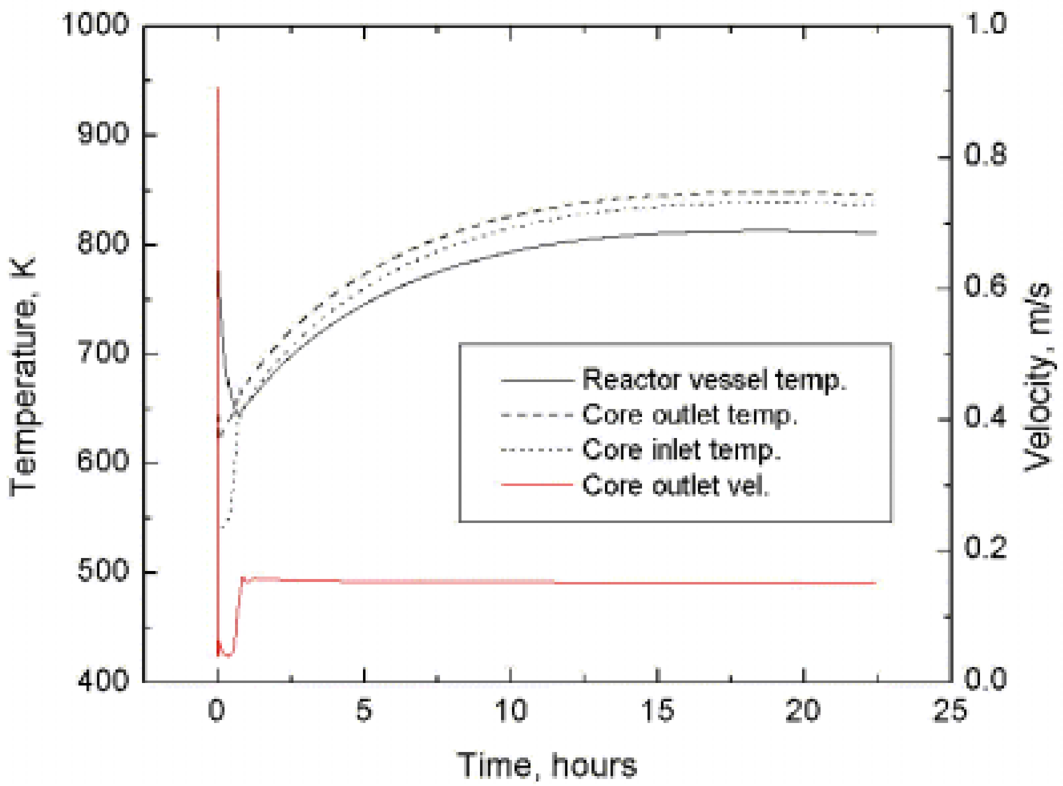

FIG. 1. STAR-CD calculated Station Blackout accident in an 800 MWt Pb/Bi cooled system with RVACS aircooling.

The SVBR75/100 design has an alternative ex-vessel emergency cooling. It consists of a large water tank that surrounds the vessel and can have an additional passive air-cooler to keep the cooling water from evaporating – the grace period without the Air Cooler would still be about 5 days [10, 16].

However, there are also some concerns about lead based coolants. One of the difficulties is the in-service inspection of the inside of the vessel. This problem is similar as in a sodium-cooled reactor. However, in new designs one could make critical components such as pumps easy to withdraw or the tube side of heat exchangers inspectable from the vessel top. Corrosion and erosion problems have been investigated for a long time in Russia that has an 80 reactor-year experience with lead/bismuth eutectic (LBE) cooling in their Alpha submarines [16]. In the EU two large laboratories have extensive experimental programs for investigating the prevention of steel corrosion by oxygen control and aluminium coating as well as thermal-hydraulics aspects of LBE coolant for the development of an experimental ADS [17]. Another concern is the generation of the Alpha emitter Po-210 and some long-lived isotopes in the LBE coolant. IPPE Obnisk proposes to re-use the coolant that is also somewhat costly. Pure lead coolant would be cleaner but it has a melting point of 327°C whereas LBE has a melting point of 125°C. The lead isotope that would become the least radioactive is Pb-206, which would, however, require isotope separation.

The LBE cooled SVBR75/100 design is complete and construction of a prototype could start soon. An international co-operation on the construction and later operation would help the participating countries to gather information about this promising technology that could potentially meet all the INPRO and GenIV criteria – regarding the economical aspects of this design. Ref. [16] gives an indication that the design cost of produced electricity in US cent/kWe-hour of this modular LBE cooled fast reactor would be 1.46, the one of a VVER-1000 2.02, of a VVER-1500 1.85, BN-1800 1.56 and steam/gas units 1.75. So far no proposed fast reactor has had a capital cost lower then that of an LWR.

3.2. Inherent safety aspects and passive approaches in gas cooled high temperature systems

In the thermal pebble bed modular reactor PBMR or the Gas-turbine Modular Helium Reactor GT-MHR the large fraction of graphite moderator makes the power density rather small. In fact it is the lowest one of all current reactor designs and makes the grace periods in accidents rather long. In a LOF or depressurisation accident the HTR core heats up and the strong negative Doppler effect brings the power down to a low power level. A hypothetical strong reactivity insertion will lead to a short –term power peak that is turned around by the prompt and negative Doppler feedback. No fission gas release from the TRISO particles is expected. Passive decay heat removal from a modular system is possible by conduction through the graphite and due to radiation to a water-cooled surface [18]. Despite these important inherent and passive features there are concerns about air ingress following depressurisation accidents and also about the subsequent cooling of the core. In a low-pressure condition there is also the possibility of water ingress because even in a direct cycle, water circuits remove about half of the heat from an HTR. A further concern is the quality assurance of some billions of coated particles per modular HTR.

A gas cooled fast reactor (GFR) has the advantage that it can breed and burn its own actinide waste, and thus, meet the sustainability criteria [19]. Inherent advantages are the possibility to inspect the vessel internals and also the proposed helium coolant is not corrosive. Disadvantages of the gas cooling are the high system pressure that can lead to depressurisation accidents and the low heat capacity and conductivity. However, both of these properties are better at the high system pressure. At this pressure the natural circulation capability in the vessel is also good enough to remove the decay heat passively. For depressurised conditions, however, forced convection is needed. Adding heavier gases like argon or nitrogen can also temporarily enhance the natural convection in a depressurised vessel [20]. In unprotected accidents the power density of the system as well as the heat resistance between fuel and gas-coolant are very important. Moreover the use of SiC coated particles adds enough moderator to soften the spectrum and to make the negative Doppler reactivity larger. published safety calculations of a GFR, with block-type fuel elements with vertical channels for the gas cooling and containing SiC-coated fuel particles and a power density of 58.2 MWt/m^3, show a rather good behaviour in unprotected reactivity transients and loss-of-flow accidents. In depressurisation accidents and assumed leaks of more than 5 cm diameter a reactor scram is needed within 53 sec. For a 10 cm diameter hole, 53.4 sec for a 20 cm hole 15.5 and 3.7 sec for a 30 cm hole are needed [21]. The core under consideration has a 10 m^3 volume, 600 MWt and a softened spectrum. It will be interesting to evaluate whether such a large fast core can be economical and whether its spectrum is still hard enough to burn plutonium and minor actinides. Other general concerns are that the gas coolant doesn't shield the structures around the core from the neutron flux and much shielding has to be installed. Moreover, disrupted or molten fuel cannot be easily cooled by gas. As for the HTR, the quality assurance of billions of coated particles is a concern. For a fast system that should meet the long-term sustainability criterion, reprocessing is also needed. This is not yet developed for SiC coated particles.

3.3. Inherent safety features of molten salt reactors

Another potential future reactor, the molten salt reactor (MSR) has also interesting inherent safety features, particularly if a pool-type design is chosen [22]. Heat-up of such a reactor leads to a level swell and since the fuel is a part of the molten salt mixture, this will reduce the reactivity and power. In order to avoid power oscillations, an overflow device could be installed. Another and more radical approach to prevent a core heat-up is a melting disc in the bottom of the pool that would release the whole salt mixture into a flat and coolable pool below [23]. However, care has to be taken to make the temperature coefficient of an MSR negative [24]. The main difference between a fast reactor and a thermal or epithermal MSR with regard to neutron economy is the much smaller critical mass. This is due to much higher fission cross section in the more thermal parts of the spectrum. In the 60's there has been a three-year operational experience with the MSR reactor in Oak Ridge that lead eventually to the resolution of all problems encountered [25].

4. DEFENSE IN DEPTH

The defense in depth approach remains the fundamental safety strategy as well as diversity and redundancy. Inherent and passive safety features are important components for these strategies. A comprehensive passive approach that is independent of the active one is needed. Such a passive approach, together with the active one that stresses the role of the operator, fulfills the diversity and redundancy strategies. In most of the defense in depth levels, inherent features and passive measures are useful. A proposal how to include them into the IAEA defense in depth levels [27] is made in Table I and a also a few examples are given:

Table I. Defense in Depth Levels [27] – the additions in italics are proposals by the authors

Levels of defence in depth	Objective	Essential means
Level 1	Preventions of abnormal operation and failures	Conservative design and high quality in construction and operation *and use of inherently safe features like low pressure coolant and negative reactivity coefficients*
Level 2	Control of abnormal operation and detection of failures	Control, limitation and protection systems and other surveillance features
Level 3	Control of accidents within design basis	Engineered safety features and accident procedures – *include passive features, e.g. pressure pulse transmitters, emergency condensers*
Level 4a	Control of severe plant conditions, including prevention of accident progression	Complementary measures and accident management - *make use of inherent and passive safety features, e.g. RVACS, large thermal inertia, passive core flooding*
Level 4b	Mitigation of the consequences of severe accidents	Complementary measures and accident management - *make use of inherent and passive safety features, e.g.* autocatalytic H_2 recombiners, *core catcher*
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

5. CONCLUSIONS

Comprehensive passive systems fully independent of the operator considerably and credibly decrease the core damage frequency and also the frequency for a large release. This is due to the independence of the two safety approaches and to the fact that operator errors can be corrected by passive systems. The use of inherent safety approaches can make the new systems considerably simpler and, thus, safer.

6. REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Terms for describing new, advanced nuclear power plants, TECDOC-872 (April, 1997)

[2] JUHN, P.E., KUPITZ, J., CLEVELAND, J., CHO, B., LYON, R.B., IAEA activities on passive safety systems and overview of international development, Nuclear Engineering and Design, 201 (2000) pp 41-59

[3] KUCZERA, B., JUHN, P.E., User Requirements for Safety of Innovative Nuclear Reactors and Fuel Cycle Installations, ICAPP'03, Paper # 3111, Cordoba, May, 2003

[4] S. JONG OH, Y. SANG CHOI, APR1400 Design: Its Safety Features and Associated Test Programs, OECD Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, France, Febr. 2002

[5] BRETTSCHUH, W., Framatome-ANP, The SWR-1000: A Nuclear Power Plant Concept with Boiling Water Reactor for Maximum Safety and Economy of Operation, ICONE9, Nice, France, April, 2001

[6] BRETTSCHUH, W., Framatome-ANP, Safety Concept of the SWR 1000, an Adavanced Boiling Water Reactor with Passive Safety Features, OECD Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, France, Febr. 2002

[7] ANS NUCLEAR NEWS September 2002, Article on Framatome ANP's SWR-1000 reactor design

[8] MESETH J., RETTSCHUH, W., Current Research on Passive Safety Devices and Advanced Fuels for Innovative Reactors, Part 1: Research on Passive Safety Devices, OECD Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, France, Febr. 2002

[9] USDOE and GENERATION IV INTERNATIONAL FORUM, A Technology Roadmap for Generation IV Nuclear Systems, GIF-002-00, Dec. 2002

[10] TOSHINSKY, G.I., et al. Safety Aspects of SVBR75/100 Reactor, OECD Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, France, Febr. 2002

[11] SPENCER, B.W., The Rush to Heavy Liquid Metal Cooling, ICONE-8, Baltimore, April 2000

[12] SEKIMOTO, H., Tokyo Inst. of Technology, S. MAKINO, Nissan Motor Co, K. NAKAMURA, Y. KAMISHIMA, Advanced Reactor Technology, Small LBE-cooled Fast Reactor for Expanding Market, ICONE10-22049, Arlington, Va, USA, 2002

[13] PlANCHEON, H.P., SACKETT, J.I., GOLDEN, G.H., SEVY, R.H., Implications of the EBR-II Inherent safety Demonstration Tests, Nuclear Engineering and Design, 101, pp 75-90

[14] WIDER, H., CARLSSON J., Jones, A.V., Lead/Bismuth-Cooled, Thorium Based ADS and Critical Systems Meet Sceptic's Criteria, ICENES 2000, Petten, Netherlands Sept. 2000

[15] CARLSSON, J., Inherent Safety Features and Passive Prevention Approaches for Pb/Bi-cooled Accelerator-Driven Systems, Doctoral Dissertation, Royal Institute of Technology, Stockolm, ISBN 91-7283, 470-6, May, 2003

[16] ZRODNIKOV, A.V., et al. FGUP Atomenergoproekt, Advanced NPPs Based on Modular Lead-Bismuth Cooled Fast Reactors, ICAPP'03, Cordoba Spain

[17] KNEBEL, J.U., FZK and R.J.M. Konings, ITU, Karlsruhe, Current and Future Research on Corrosion and Thermalhydraulics of HLM Cooled Reactors and on LMR Fuels for Fast Reactor Systems, OECD Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, France, Febr. 2002 - also: http://www.kalla.fzk.de/

[18] KUGLER, K., BARNERT, H, . PHLIPPEN, P.W., SCHERER, W., FZ JÜLICH, Safety of HTR – State of Knowledge and Necessary Research, OECD Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, France, Febr. 2002
[19] NEWTON T.D., Serco Assurance, UK, Flexibility of the Gas Cooled Reactor to Meet the Requirements of the 21st Century, Reactors and Fuel Cycle Installations, ICAPP'03, Cordoba, May, 2003
[20] GARNIER, J.C., POETTE, C., MATHIEU, B., CONTI, A., GAILLARD, J.P., CEA-Cadarache, Preliminary Design of an Advanced Gas Cooled Fast Reactor Core, Fuel Forms and Primary System Concept, ICAPP'03, Cordoba, May, 2003
[21] BASSI, C., CEA-Cadarache, Core Transient Behaviour of the Gas Cooled Fast Reactor, ICAPP'03, Cordoba, May, 2003
[22] VASILIEV, A., et.al., Kurchatov Institute, Conceptual Design of Molten-Salt Reactor-Burner of Minor Actinides, Physor 2002, Seoul, Korea
[23] VERGNES, A., GARZENNE, C., LECARPANTIER, D., MOUNEY, H., DELPECH, M., CEA- Cadarache, The Amster Concept: A Configuration Generating its own Uranium with a Mixed Thorium and Uranium Support, Global 2001, Paris Sept. 2001
[24] CARPENTIER, D., GARZENNE, C, HEUER, D., NUTIN, A., IN2P3, UJF Grenoble, Temperature Feedbacks of a Thermal Molten Salt Reactor: Compromise between Stability and Breeding Performance, ICAPP'03, Cordoba, May 2003.
[25] MACPHERSON, H.G., ORNL http://www.moltensalt.org/mSR_Adventure.html
[26] CARELLI, M.D., WINTERS, J.W., CUMMINS W.E., BRUSCHI, H.J., Safety Features and Research Needs of Westinghouse Advanced Reactors, OECD Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, France, Febr. 2002
[27] IAEA, INTERNATIONAL NUCLEAR ADVISORY BOARD (INSAG), Basic Safety Principles of Nuclear Power Plants, IAEA Safety Series No. 75-INSAG-3, Vienna, 1988

RUSSIA AND INNOVATIVE PROJECTS INPRO AND GENERATION-IV

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Abstract. The report provides a brief comparative analysis of the Generation-IV and INPRO contents, outlines the pathways for interaction and rapprochement between them, and suggests the possible roles of the U.S. and Russia in this process. Upon further evolution of the INPRO, one can easily anticipate that both programs would have several common standpoints regarding both, the requirements to innovative reactors and fuel cycles (this is already evident), and the concepts and approaches selected for further development and deployment. All this will create the prerequisites for this or that degree of future rapprochement of the two programs. The first step here might be improvement of the coordination between these two projects, with Russia joining the Generation-IV and the U.S. joining INPRO within the next step. To secure an option of such cooperation there is a need in joint actions targeted at mitigation or resolution of the observed contradictions preventing Russia from joining the Generation-IV and the U.S. from joining the INPRO. One of possible pathways to the resolution is associated with working out, co-jointly, proposals on the construction in several countries of the NPPs with reactors providing a better degree of resistance to the proliferation of nuclear weapons.

1. INTRODUCTION

Among innovative energy technologies, nuclear power is a prime candidate that, in view of its technological and infrastructural readiness, may become a major sustainable energy leader of the 21st century. Today in many countries where nuclear power is used, the programs are being pursued to improve nuclear power's performance with respect to economic, safety, waste, and nonproliferation criteria, to draw an outline of future nuclear power that would be acceptable both to the stakeholders and consumers, and to advance the necessary RD&D. International cooperation is being pursued for this purpose, showing significant progress along several important trends.

Russia has initiated international cooperation on innovative nuclear power technologies within the INPRO project performed under the auspices of IAEA. Phase one of the INPRO, under way currently, is dedicated to the selection of criteria and development of the methodology to compare different innovative concepts. Time framework for implementation of projects and technologies under the INPRO is anticipated to cover the nearest fifty years.

The US facilitate international cooperation on innovative reactors and fuel cycles within the Generation-IV International Forum. As of today, identified are six most promising reactor and fuel cycle concepts that could be deployed by 2030, and the R&D areas necessary to advance these concepts for potential commercialization. In 2003, the R&D on at least four of the identified reactor system and fuel cycle concepts were started on a bilateral and multilateral bases.

Both these programs are targeted at identification and subsequent advancement of the innovative nuclear power technologies that could resolve principal issues of humankind sustainable development and, thus, could become basic in the evolving nuclear power of the 21st century. They progress independently nowadays, though avenues for the effective collaboration and, probably, coordination between them are being pursuit, the more so as a priori it is clear that they might complement and enrich each other

Presidents Putin and Bush indicated priority areas for collaboration in their joint statement at the US-Russian Summit that took place in Moscow and St. Petersburg in May, 2002, calling for collaboration in the research and development of new, more environmentally safe nuclear power technologies.

Within the context of this statement, both, the authorities and independent technical experts of the U.S and Russia have developed proposals on collaboration in this field between the two countries. For example, a plan of interactions and joint activities providing for Russia to join the Generation-IV International Forum and, later on, for the U.S. to join the INPRO was set forward in the "Joint Report of the United States and Russian Federation Experts Group on Advanced Nuclear Technologies" [1].

This report provides a brief comparative analysis of the Generation-IV and INPRO contents, outlines the pathways for interaction and rapprochement between them, and suggests the possible roles of the U.S. and Russia in this process.

2. COMPARISON OF THE GENERATION-IV AND INPRO INTERNATIONAL PROGRAMS

2.1. Brief characterization of the Generation-IV program

Generation-IV is a program that refers to the development and demonstration of one or more Generation-IV nuclear energy systems that offer advantages in the areas of economics, safety and reliability, sustainability, and could be deployed commercially by 2030 [2].

Being initiated by the U.S. DOE, the Generation-IV has made a start-up in U.S.A. in 2000, soon gained an international status through the Generation-IV International Forum and, with time, is expected to evolve into a full-scale international R&D project.

The Generation-IV steps so far completed are:

— Definition of the Technology Goals;
— Determination of how to measure concepts against the Goals, i.e., evaluation methodology;
— Identification of the reactor technology concepts to undergo initial evaluation;
— Detailed concept information gathering;
— Preliminary evaluation of concepts against the Generation-IV Goals;
— Concept screening and selection of the 6 to 8 most promising reactor and fuel cycle concepts;
— Identification and establishment of structured R&D plans to guide development of the most promising concepts for deployment by 2030-2040.

Six most promising nuclear energy system concepts have been selected, and the Roadmap for their further advancement identifying technology gaps and providing the relevant structured R&D plans has been developed. Each of the six system concepts appears as a combination of reactor and fuel cycle options, complete from uranium mining to final disposal of waste.

R&D on at least four of the selected system concepts have been started and are in progress. They are performed on a national, bilateral, and multilateral bases.

It should be noted that R&D for advanced fuel cycles are not the subject of the Generation-IV international program and are carried out separately, within the U.S. DOE Advanced Fuel Cycle Initiative (AFCI).

Bilateral and multilateral cooperation within the Generation-IV is facilitated by the Generation-IV International Forum (GIF). Within GIF programs, each member controls its own research management process, any costs arising from the activities contemplated by GIF are borne by the member that incurs them.

Currently, the members of GIF are: Argentina, Brazil, Canada, France, Japan, Republic of Korea, South African Republic, Switzerland, Great Britain, and the United States.

2.2. Brief characterization of the INPRO project

In 2001, Russia has initiated international collaboration on innovative nuclear power technologies within the INPRO project [3] performed under the auspices of IAEA.

The objectives of INPRO, as defined in the Terms of Reference, are [3]:

— To help to ensure that nuclear energy is available to contribute in fulfilling, in a sustainable manner, energy needs in the 21st century;
— To bring together all interested Member States, both technology holders and technology users, to consider jointly the international and national actions required to achieve desired innovations in nuclear reactors and fuel cycles that use sound and economically competitive technology, and are based – to the extent possible – on systems with inherent safety features and minimize the risk of proliferation and the impact on the environment;
— To create a process that involves all relevant stake holders that will have an impact on, draw from, and complement both the activities of existing institutions and ongoing initiatives at national and international levels.

In the initial phase (Phase 1A), which was performed mainly in 2002, work proceeded in five subject areas recognized as important for the future development of nuclear energy technology, and was devoted to the selection of criteria and development of methodologies and guidelines for the comparison of different concepts and approaches, taking into account the compilation and review of such concepts and approaches; and determination of User Requirements in the subject areas.

Simultaneously, started was the preliminary collection and analysis of innovative nuclear energy approaches and technologies as made available by Member States.

In 2002 - 2003, works for Phase 1A of the INPRO produced sets of User Requirements in the subject areas and methodology for the comparison of different concepts and approaches. The degree of completion still varies between particular tasks.

In 2003 it is planned to complete Phase 1A and move on to Phase 1B which would include the evaluation of several innovative technologies by Member States, and also preliminary collection of the information on innovative reactors and fuel cycles. Particular communications and interactions with other national and international stakeholders, such as OECD/NEA and Generation IV International Forum (GIF), will be continued.

In 2003, it is also planned to collect the data on innovative nuclear energy systems from Member States. Further on, the assessment of innovative nuclear reactor and fuel cycle concepts against the requirements and criteria selected would be carried out, with active participation of Member States in this process being envisaged also.

Upon successful completion of the first phase, a second phase of INPRO may be initiated. Drawing on the results from the first phase, it will be directed to:

— Examining in the context of available technologies the feasibility of commencing an international project;
— Identifying technologies, which might be appropriate for implementation by Member States of such an international project.

The time frame for realization of the INPRO projects and technologies would encompass the nearest 50 years.

As of October 2002, the European Commission and 12 Member States, namely, Argentina, Brazil, Canada, China, Germany, India, Republic of Korea, Russian Federation, Spain, Switzerland, the Netherlands and Turkey were participating in the INPRO.

2.3. Common points and differences

The Generation-IV Technology Goals and so far developed User Requirements of the INPRO are collated in Table I.

Analysis of the overall contents of these two international projects makes it possible to draw the following conclusions:

(i) The technology goals of Generation-IV and User Requirements by the INPRO have many similar or even identical statements. More often this is the case for issues of safety, environment, fuel cycle and waste, proliferation resistance, and sustainability;
(ii) The final structure and contents of User Requirements within the INPRO might be essentially more detailed and thorough as compared to those of the Generation-IV Technology Goals. In particular, component-specific User Requirements have already been developed for several subject areas;
(iii) Generation-IV addresses mostly the demands of a few industrially developed countries, while INPRO is biased for a more in-depth consideration of nuclear power use in developing countries, for the assessment of regional, institutional, and infrastructural aspects of a global nuclear power organization. In this context, of special value may be INPRO's User Requirements for cross-cutting issues;
(iv) While Generation-IV narrows the scope of its consideration to the 'separate' nuclear energy systems with reactors of different types, consideration of the combinations of such systems, i.e., of the scenarios of nuclear power development at the national, regional, and global levels is provided for by the INPRO;
(v) As of today, both Generation-IV and INPRO appear not to make a duly consideration of the issue of materials disposition in application to innovative nuclear energy

systems. However, it seems that this issue will remain crucial for the time being, while the capability of innovative nuclear energy systems to deal with newly released weapons materials in an efficient and secure way could be considered as one of the criteria for screening and final selection;

(vi) The qualitative methodological approaches as developed within the INPRO and Generation-IV for the assessment of candidate innovative concepts differ in that the Generation-IV is targeted at screening and down-selection, while INPRO is targeting more on accepting or not accepting;

(vii) While Generation-IV is already in the phase of initial R&D, INPRO appears not to be completely through with User Requirements formulation.

Table I. Generation-IV Technology Goals vs. INPRO User Requirements [2,3]

Generation-IV [2]	INPRO [3]
Sustainable development	
— Sustainability is the ability to meet the needs of the present generation while enhancing the ability of future generations to meet society's needs indefinitely into the future	— Development that meets the needs of the present without compromising the ability of future generations to meet their own needs
— Generation IV nuclear energy systems including fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production — Generation IV systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for public health and the environment — Sustainability requires the conservation of resources, protection of the environment, preservation of the ability of future generations to meet their own needs, and the avoidance of placing unjustified burdens upon them	**Resources, demand** — Learning rates as defined by the experience and RD&D programmes, possible scenarios of minimum and maximum nuclear power growth *Environment, fuel cycle, and waste* — An integral approach to the assessment of environmental impacts: all important material and energy flows in, out, and through the system should be accounted for, as well as all relevant factors (sources, stressors, pathways, receptors and endpoints) — The net energy output of the system should exceed the energy required to implement and operate the system within an acceptably short period of time, to be determined by stakeholder consensus — Proven reserves and inventories of important materials, such as uranium, thorium and plutonium, should be sufficient to sustain the system for at least 100 years — The expected (best estimate) environmental effects of the system should be within the performance envelope of the life cycle of current nuclear power technology

Table I. (continued)

Generation-IV	INPRO
— Having a positive impact on the environment through the displacement of polluting energy and transportation sources by nuclear electricity generation and nuclear-produced hydrogen — Allowing geologic waste repositories to accept the waste of many more plant-years of nuclear plant operation through substantial reduction in the amount of wastes and their decay heat — Greatly simplifying the scientific analysis and demonstration of safe repository performance for very long time periods (beyond 1000 years), by a large reduction in the lifetime and toxicity of the residual radioactive wastes sent to repositories for final geologic disposal — Extending the nuclear fuel supply into future centuries by recycling used fuel to recover its energy content, and by converting ^{238}U to new fuel	*Waste management* — For each waste in the energy system, a permanently safe end state should be defined, [with] any release of material to the environment, normalized to unit of energy produced, [to] be below that which is acceptable today — Waste management systems should be designed to assure that their associated radiological effects on humans are below the levels acceptable today — Waste management strategies should be such that the adverse environmental effects from all parts of the energy system and the complete life cycle of facilities are optimized — The energy system should be designed so as to minimize the production of wastes and particularly the production of wastes containing long-lived toxic components that would be mobile in a repository environment — All costs associated with the management of the wastes should be reliably estimated and internalized as part of the overall cost of the energy system
Reliability and safety	
— Generation IV nuclear energy systems operations will excel in safety and reliability — Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage — Generation IV nuclear energy systems will eliminate the need for offsite emergency response — Enhancing public confidence in the safety of nuclear energy	Future nuclear reactors and fuel cycle installations shall: — be so safe that they can be sited in locations similar to other industrial facilities — have a lower risk associated with fuel damage than current plants — prevent releases of radioactivity that could necessitate evacuation of people living nearby — ensure safety to people and the environment of the whole fuel cycle

Table I. (continued)

Generation-IV	INPRO
Economics	
— Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources — Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects — Allowing the distributed production of hydrogen, fresh water, district heating, and other energy products to be produced where they are needed	**Resources, demand, and economics** — Minimum and maximum growth rates to learn from, forecasts of energy production and consumption scenarios, converting learning rate targets to static cost targets, etc.
Non-proliferation and physical protection	
— Generation IV nuclear energy systems including fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials — Providing continued effective proliferation resistance of nuclear energy systems through the increased use of intrinsic barriers and extrinsic safeguards — Increasing physical protection against terrorism by increasing the robustness of new facilities	**Proliferation resistance:** — Proliferation resistance intrinsic features and extrinsic measures help ensure that future nuclear energy systems will continue to be an unattractive means to acquire materials for a nuclear weapons programme — Proliferation resistance could be enhanced when complementary and redundant features and measures provide defense in depth — Proliferation resistance will be enhanced when taken into account as early as possible in the design and development of a nuclear energy system — Effective use of intrinsic proliferation resistance features facilitates efficient application of extrinsic measures — Extrinsic proliferation resistance measures, such as control and verification measures, will remain essential, whatever the level of effectiveness of intrinsic features — From a proliferation resistance point of view, development and implementation of intrinsic features should be encouraged

Table I. (continued)

Generation-IV	INPRO
	Physical protection — Started in 2003, no information by the time this paper was prepared
Cross-cutting issues	
— Confidence in the safety of nuclear power needs to be increased. Of all the goal areas, those regarding safety of nuclear energy systems, protection of nuclear materials and facilities within the system against acts of terrorism, and nuclear proliferation are most closely linked to public confidence in nuclear energy — The opportunities for advancing Generation IV systems will also depend on gaining public confidence, which can be enhanced through the openness of the process of developing and deploying Generation IV systems. The findings of Generation-IV Roadmap and R&D plans that are based on it will be communicated to the public on a regular basis, and opportunities for stakeholder groups to provide feedback on the plans will be offered	*Socio-political infrastructure:* — It should be demonstrated that innovative nuclear concepts are responding to the concerns about safety, waste and proliferation as contribution to the improvement of public acceptance of nuclear power — The main concerns on safety, waste and proliferation, that are influencing the public acceptance have to be responded by the world wide application of the top level requirements . Countries should cooperate in reaching this goal — A general requirement in relation with public acceptance should be the communication between the public and other stakeholders on the scenarios and choices in energy supply. The implementation of nuclear programs and the performance of the nuclear power structures should also be subject of open communication

Table I. (continued)

INPRO
Legal and institutional infrastructure: — A license for the design and application of innovative reactor systems and fuel cycles needs to be accepted by any user country that wants to use these systems — Licenses for innovative reactors should be based on the top-level requirements. These requirements should be internationally agreed — A requirement for the deployment of innovative nuclear concept may be the establishments of international or regional nuclear authorities and inspection bodies — Condition for the growth of international operating companies is that the insurance of risk attributed to the use nuclear power can be handled on the same way as other industrial risks. The liability issue has to be reviewed as market structures are changing
Economic and industrial infrastructure: — Companies involved in research, development and supply of nuclear technology should develop towards global companies — Market demand and the specific needs have to be recognized by the developers — The nuclear power structure could be optimized if components in different countries will be part of an international multi-component system that could respond to the demand for sustainability, including the final storage of waste
Human resources and knowledge: — Given the need for human resources and knowledge as a base for the development of innovative nuclear concepts and fuel cycles international cooperation in this field should be enhanced

3. PATHWAYS FOR COOPERATION BETWEEN GENERATION-IV AND INPRO, AND POSSIBLE ROLE OF RUSSIA IN THIS PROCESS

The meetings held by the Policy Group of the Generation-IV International Forum on October 10-11, 2001 and March 14-16, 2002 produced a statement and a resolution on the importance of Russian participation in Generation-IV. In particular, it was agreed that Russian membership to the GIF is desirable as early as feasible, and even before the final down-selections are made. There is no doubt that such decisions are backed by clear understanding of the fact that Russian technology experience in many selected priority trends surpasses the relevant experience of GIF-member countries. First of all, this may be attributed to the advanced technologies of nuclear fuel cycle as linked to the use of non-aqueous processes of fuel reprocessing, and also to the developments in the field of advanced fast reactors with various coolants (sodium, heavy metals, helium).

Russian nuclear industry has accumulated significant technological background along all trends selected within the Generation-IV. This statement is clearly illustrated by the data of Table II. Namely this lies behind the Resolution of Generation-IV International Forum of 2002 welcoming Russian participation at as early as possible stage.

At the same time, Russia joining the Generation-IV would not provide a desirable level of their cooperation and coordination. The reason is that Generation-IV international program was initiated by the U.S.A. and pursues mostly the U.S. interests. Its scope of problems matches the demands of only a few leading countries, the issues of sustainable global development and energy demands of the developing countries are left out. In this context, IAEA shall unconditionally go on with the advancement and promotion of the INPRO.

Different from Generation-IV, the initiated by Russia INPRO international project was initially targeted at the interests of global community. Hence, the provided more profound level of consideration for global and regional aspects of nuclear power, including the spectrum of possible solutions such as fuel and NPP leasing, multi-national ownership of nuclear fuel cycle centers, etc. It is anticipated that INPRO would involve a more broad spectrum of the technology proposals for innovative reactors and nuclear fuel cycles, the spectrum that would meet the demands of nearly all countries in the world, including not only those that are nuclear stakeholders, but also those that intend to be users only. Proposals for INPRO may include different options for the advanced light water reactors, a variety of nuclear batteries for autonomous energy supply, proposals on fast-breeder reactors ,thorium fuel reactors, etc.

Upon further evolution of the INPRO, one can easily anticipate that both programs would have several common standpoints regarding both, the requirements to innovative reactors and fuel cycles (this is already evident), and the concepts and approaches selected for further development and deployment. All this will create the prerequisites for this or that degree of future rapprochement of the two programs. The first step here might be improvement of the coordination between these two projects, with Russia joining the Generation-IV and the U.S. joining INPRO within the next step.

In addition to this, it is very important to avoid a new “schism” between the U.S. and others with significant nuclear energy deployments (e.g., Britain, France, Russia, Japan,…). The “schism” of the 1970’s concerning the recycling of plutonium inhibited fuel cycle development, especially with respect to development of next generation technologies that could better address the concerns of newly released military plutonium disposition [4].

When evaluating the issue of possible interaction between the Generation-IV and INPRO from today’s perspective, it is worthwhile to draw attention to the opinion of independent Russian and U.S. experts who suggest that the degree of convergence between these parallel efforts should be discussed in the pursuit of Russia-U.S. collaboration [4].

The “Joint Report of the United States and Russian Federation Experts Group on Advanced Nuclear Technologies” [1] presents a specific plan of interactions between the U.S. and Russian sides providing for Russia to join the Generation-IV International Forum and, later on, for the U.S. to join the INPRO. If being realized successfully, such type of coordination between the two programs could be started as early as in 2003. However, the realization of this specific plan was tied to the resolution of a political problem currently present in the U.S.-Russian relations.

To secure an option of such cooperation there is a need in joint actions targeted at mitigation or resolution of the observed contradictions preventing Russia from joining the Generation-IV and the U.S. from joining the INPRO. In turn, negligence may result in a new “schism” between the U.S. and Russia. One of possible pathways to the resolution is associated with working out, co-jointly, proposals on the construction in several countries of the NPPs with reactors providing a better degree of resistance to the proliferation of nuclear weapons.

Finally it should be mentioned that joining efforts of the world community in development and deployment of the innovative nuclear technologies will make it possible to change the slogan "Atoms for Peace" to "Atoms for the prosperity".

Table II. The Generation-IV Selections and Russian Technological Experience

Generation-IV [2] (October 2002)	Russia
Reactors with thermal neutron spectrum	
Supercritical Water-Cooled Reactor System SCWR	
— 1700 MW (el.) — 500-550 °C at core outlet — 25 MPa operating pressure — 44% efficiency — Thermal and fast neutron spectrum options — Open and closed fuel cycle options, advanced aqueous reprocessing methods	— 20 years of RD&D, including many experimental studies on thermal-hydraulics of the supercritical coolant — Design and development of thermal and fast supercritical reactor projects, including reactors with spectrum shift, within the unit power range from 100 to 1800 MW (el.) — Works stopped in early 1990ies due to insufficient funding — aqueous methods of reprocessing are mastered and commercially operated
Very-High-Temperature Reactor System VHTR	
— 600 MW (thermal) — 1000 °C at core outlet — A direct Brayton cycle gas turbine cycle for high thermal efficiency — Helium-cooled core based on either the prismatic block fuel of the Gas Turbine - Modular Helium Reactor (GT-MHR) or the pebble fuel of the Pebble Bed Modular Reactor (PBMR) — A once-through uranium cycle — The VHTR system is primarily aimed at relatively faster deployment of a system for high-temperature process heat applications, such as coal gasification and thermochemical hydrogen production, with superior efficiency — Electricity co-production is optional	— 50 years of experience in RD&D for HTGR reactors of different types and destinations; — Several technical projects completed (ABTU-50, VG-400 with 950 °C outlet temperature) — A prototype of very high temperature reactor with 3000 K outlet temperature was operated — A tremendous volume of experimental verification has been performed for all trends — GT-MHR international project is being developed — Cooperation with South African Republic on PBMR — 30 years of experience in the development of technological processes for the production of hydrogen, conversion of the organic fuels, etc.
Molten Salt Reactor System MSR	
— The fuel is a circulating liquid mixture of sodium, zirconium, and uranium fluorides — 1000 MW (el.) — High core outlet temperatures (700-800 °C) — A full actinide recycle fuel cycle is envisioned — Envisioned for missions in electricity production and waste burndown	— 30 years of RD&D for fluoride technologies; physicochemical studies of molten salt fuel, loop and in-pile tests, etc. — Volatile fluoride reprocessing method mastered at a lab scale, verified at pilot scale — Conceptual proposal for molten salt reactor – SNF & HLW burner

Table II. (continued)

Reactors with fast neutron spectrum	
Sodium-Cooled Fast Reactor System SFR	
— An intermediate size (150 to 500 MWe) sodium-cooled reactor with a uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in collocated facilities — A medium to large (500 to 1500 MWe) sodium-cooled fast reactor with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors — A full actinide recycle fuel cycle is envisioned	— 50 years of RD&D, several prototype reactors — 20 years of operation of the BN-350 reactor with water desalination circuit — Commercial plant with BN-600 sodium cooled fast reactor is being operated (as of today, the only one in the world) — The NPP with an advanced sodium cooled fast reactor BN-800 is under construction — Preliminary project of the BN-1600 reactor has been developed, works stopped in 1992 — A tremendous volume of RD&D has been carried out, including experimental verifications for both, traditional oxide fuel, and new high density fuels (such as metallic, nitride, and carbide) — The technology of volatile fluoride reprocessing and the pyroprocess for the chloride salts have been verified up to the stage of pilot facilities, works are being continued
Lead-Cooled Fast Reactor System LFR	
— Lead or lead-bismuth coolant — A battery of 50–150 MWe that features a very long refueling interval, a modular system rated at 300–400 MWe, and a large monolithic plant option at 1200 MWe — The most advanced of these is the Pb-Bi battery, which employs a small size core with a very long (10–30 year) core life. The reactor module is designed to be factory-fabricated and then transported to the plant site. — Metallic or nitride fuel — Natural circulation — Core outlet temperature 550 ^{o}C, may be increased up to 800 ^{o}C with the use of advanced structural materials — Nuclear battery can be used to produce hydrogen or desalinated water — A full actinide recycle fuel cycle with central or regional fuel cycle facilities is envisioned.	Lead-bismuth reactors: — 30 years of RD&D, operated were the prototypes of navy reactors, the total operation period is 80 reactor-years Lead cooled reactors: — 20 years of RD&D — Technical project for the demonstration unit of 300 MW (el.) reactor - BREST-300 - is being developed — Conceptual proposal for BREST-1200 reactor has been developed Small reactors: — 40 years of R&D. Hundreds of operated and now operating navy installations — About 20 proposals / projects for light water and lead-bismuth reactors

Table II. (continued)

Gas-Cooled Fast Reactor System GFR	
— Helium coolant — Core outlet temperature 850 °C — A direct Brayton cycle gas turbine cycle for high thermal efficiency — Closed fuel cycle and excellent performance and actinide management — Composite ceramic fuel, advanced fuel particles, or ceramic clad elements of actinide compounds — Basic missions are electricity production and actinide management, although it may be able to also support hydrogen production	— 20 years of R&D, several projects of NPPs with fast gas cooled reactors — Works on these projects stopped in early 1990ies — Verified technologies of the ceramic and coated particle fuel, including the non-traditional ones — High temperature volatile fluoride method of fuel reprocessing verified up to the stage of pilot prototype creation — Direct gas-turbine cycle being developed within the GT-MHR project — 30 years of experience in the development of technological processes for the production of hydrogen, conversion of the organic fuels, etc. — Production of hydrogen and process heat was thoroughly investigated in application to HTGR (VG-400)

4. REFERENCES

[1] JOINT REPORT OF THE UNITED STATES AND RUSSIAN FEDERATION EXPERTS GROUP ON ADVANCED NUCLEAR TECHNOLOGIES, Minatom-DOE, July 31, 2002

[2] GENERATION-IV files at: http://gen-iv.ne.doe.gov

[3] THE INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES REPORT. Draft #0 of Phase-1A. IAEA, December 24, 2002

[4] FUTURE OF NUCLEAR POWER: ENERGY, ENVIRONMENT, AND SECURITY, Materials of the U.S. – Russian Workshop, Carnegie Endowment for World Peace, Moscow, July 22-24, 2002. Atomnaya Energia, November 2002 , vol. 93, issue 1, p. 327-342

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