PROCEEDINGS

International Seminar held in Cairo, Egypt, 27–31 May 2001
Organized by the International Atomic Energy Agency
in co-operation with the OECD Nuclear Energy Agency and the
World Nuclear Association

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Energy For Sustainable Development

Small and Medium Sized Reactors: Status and Prospects

INTERNATIONAL ATOMIC ENERGY AGENCY

The originating Section of this publication in the IAEA was:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

SMALL AND MEDIUM SIZED REACTORS:
STATUS AND PROSPECTS
IAEA, VIENNA, 2002
IAEA-CSP-14/P
ISBN 92-0-114802-X
ISSN 1563–0153

Printed by the IAEA in Austria
August 2002

FOREWORD

During the early years of nuclear power deployment, the plants entering service were dominated by what are now considered small (less than 300 MW(e)) and medium (300 to 700 MW(e)) reactors. In the late 1970s, the balance shifted to larger size plants to serve the requirements of industrialized countries. However, since the early 1990s, the increased interest of developing countries in nuclear power, mainly in Asia, has resulted in intensified efforts in development of small and medium sized reactors (SMRs). Also, in industrialized countries, electricity market deregulation is calling for power generation flexibility that SMRs may offer. Apart from electricity generation, SMRs are of particular interest for non-electrical applications of nuclear energy, such as desalination of seawater and district heating. In recognition of the current global interest in small and medium sized reactors this seminar was convened to provide a forum for the exchange of information by experts and policy makers from industrialized and developing countries on the technical, economic, environmental, and social aspects of SMR development and implementation in the 21st century, and to make this information available to all interested IAEA Member States. Keynote addresses also provided information on global energy demand and supply and international trends impacting the development and introduction of SMRs.

The seminar was organized by the International Atomic Energy Agency (IAEA) in co-operation with the OECD Nuclear Energy Agency and the World Nuclear Association. It was hosted by the Egyptian Nuclear Power Plants Authority on behalf of the Government of Egypt.

Two hundred forty seven attendees from 39 countries and 5 international organizations participated in the seminar. The majority of the participants were from developing countries.

The seminar not only provided valuable up to date information on SMRs, it also highlighted the importance of continued international co-operation in the development and application of nuclear power for peaceful uses throughout the world.

These proceedings contain the papers submitted by the authors, summaries of two panel discussions and also include the opening and closing addresses.

The IAEA wishes to thank the authors, panelists, session chairs, and participants for their contributions in making a successful seminar. The members of the Steering Committee who participated in two Advisory Group meetings in Vienna are also gratefully thanked.

The IAEA gratefully acknowledges the support and generous hospitality extended to the seminar by the Government of Egypt and the Nuclear Power Plants Authority. The IAEA greatly appreciates the many activities of the Local Organizing Committee led by S.B. Abdel Hamid of the Nuclear Power Plants Authority of Egypt. The IAEA acknowledges the assistance and support of K. Hesketh in the compilation of the papers for publication.

The IAEA technical officer responsible for this publication was D. Majumdar of the Division of Nuclear Power.

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SUMMARY

At the end of 2000, 438 nuclear power plants were operating in 30 countries with a total capacity of 351 GW(e). The global nuclear share of electricity was 16.1%. Much of the nuclear share is concentrated in industrialized countries, but a number of developing countries have already deployed nuclear power projects and some are considering doing so. During the early years of nuclear power deployment, in the 1950s and 1960s, the plants were dominated by what are now considered small (less than 300 MW(e)) and medium-sized (300 to 700 MW(e)) reactors. Then in the 1970s and 1980s many large reactors (700 MW(e) to 1500 MW(e)) were constructed. However, since the early 1990s, the interest of developing countries, mainly in Asia, has resulted in increased efforts on the design of small and medium sized power reactors. Also, in industrialized countries, electricity market deregulation is calling for power generation flexibility that smaller reactors may offer. Small and medium reactors (SMRs) are also of particular interest for non-electric applications such as seawater desalination and district heating, and, in the future, hydrogen production. In the next 50 years, electricity demand is expected to be tripled, most of which will come from developing countries. In light of this, the IAEA organized the International Seminar on Status and Prospects for Small and Medium Sized Reactors in Cairo, Egypt from 27–31 May 2001. The objective of the seminar was to provide a forum for the exchange of information by experts and policy makers from developed and developing countries on the technical, economic, environmental, and social aspects of SMR development and implementation in the 21st century, and to make this information available to all interested Member States.

Two hundred forty-seven attendees from 39 countries and 5 international organizations participated at the seminar. One hundred and eighty nine of these came from developing countries and 45 from developed countries with 13 from the international organizations.

The seminar and the accompanying exhibit were opened by a distinguished group, which included the Minister of Electricity and Energy of Egypt and the Director Generals of the IAEA, OECD/NEA and the World Nuclear Association. The Egyptian Minister conveyed the message on nuclear energy for the developing countries on behalf of the Prime Minister of Egypt, which emphasized: (1) long-term commitment of the country for nuclear energy, (2) transfer of nuclear technology from the developed world, (3) the importance of and commitment to safety, (4) reduction of the high cost of development and investment, (5) dealing with the waste and back-end fuel-cycle issues and (6) enhancement of public acceptance.

IAEA Director General M. ElBaradei, said that innovation, adaptability, and collaboration are keys to success for the SMRs. With the advent of telecommunications and the global marketplace, the world has become a much smaller place, and the demand for a higher standard of living is increasing everywhere — yet an estimated two billion people still lack access to electricity. Dramatic increases in electricity demand are expected over the next several decades — with the growth rate in developing countries expected to be three times faster than in industrialized countries. To meet this growth — inevitable for economic and social development — he contended that a total reliance on fossil fuels and large hydroelectric facilities is not sustainable, and an expanded future role of nuclear power must be considered. ElBaradei said the future of nuclear power depends upon success in meeting four basic challenges. The first challenge is to develop clear national and international strategies for the disposal of high level nuclear and radioactive waste. The second challenge is to remain vigilant in ensuring the continued safety of operations at nuclear facilities. The third challenge

involves outreach to civil society — engaging the public and decision makers in a fair evaluation of the relative merits of the different energy options. The fourth challenge entails the development of new, innovative reactor and fuel cycle technologies. To be successful, these new technologies should incorporate inherent safety features, proliferation resistant characteristics, and reduced generation of waste. They must also be capable of generating electricity at competitive prices while satisfying both regulators and investors.

OECD/NEA Director General L. Echavarri emphasized that the beginning of the 21st century is characterised by significant changes in the energy policy-making framework of most countries. Globalisation of the world economy, deregulation of electricity markets, privatisation of the electricity sector, increasing concerns about the need to protect the environment and awareness of sustainable development goals are among the major trends affecting policy making and decisions in the energy sector. All those factors have impacts on nuclear energy programmes and may affect SMR development in particular. National energy policies are based upon country specific contexts and priorities but the main driving factors in energy policy making are similar world-wide. The evolution of energy supply mixes and the rate of change between alternative sources or technologies have been driven by a limited number of factors relating to economic development and competitiveness, as well as social and environmental protection issues. Although recent trends place emphasis on market mechanisms to ensure competitiveness, governments, especially in OECD countries, are increasingly considering an integrated approach to policy making, within a sustainable development framework, incorporating economic, social and environmental dimensions.

WNA Director General J.B. Ritch III announced the new charter of the World Nuclear Association (formerly Uranium Institute) as an organization representing the nuclear industries and that its membership is being enlarged. It will provide a global nuclear forum and a commercial and technical meeting place for those engaged in nuclear power. He very aptly said that Egyptians as a nation and people stand at the bridge-point between the many separations in our world — between past and future, North and South, East and West, poor and wealthy, developed and developing–and in the realm of nuclear affairs, between those nations with nuclear energy and those without. So, it was particularly appropriate that a seminar on the future of nuclear power was held in Cairo.

The seminar was organized in a series of 13 sessions, which included 3 panel discussions and contained a total of 82 speakers in addition to the 4 opening addresses. The panels discussed challenges of SMR deployment, incentives for introduction of SMRs in developing countries and solutions leading to increased deployment of SMRs. Separate sessions dealt with economics and financing, non-proliferation fuel cycles issues, reactor designs and applications.

A significant item, stressed at the very beginning, was that the population growth of the world has been decelerating since 1990, and the world population may barely reach 8 billion around 2050 and it may start to decline shortly thereafter; virtually all of the 21st century's population growth will occur in developing countries. The population will concentrate around cities and the number of Mega Cities with 15 million or more people will increase from 5 to 15 and all in developing countries. It was said that if nuclear energy cannot play a role in the developing countries, it is destined to be a sideshow in the global energy picture.

Competitiveness remains a cornerstone in energy policy making, but the framework within which comparative economic assessments are conducted is evolving. Increasing emphasis is placed on market mechanisms for promoting optimised energy supply mixes, in particular for electricity generation. The key factors affecting the economic competitiveness of alternative electricity generation sources and technologies are essentially fossil fuel prices, capital costs, expected rates of return on investments, and technological performance, e.g. thermal efficiency, availability factor and technical lifetime of power plants. In particular, combined cycle gas turbines, which are the main competitor for SMRs at present, may lose most of their competitive margin if gas prices continue to rise.

Regarding economic and financial aspects, total capital cost is lower for SMRs than for large size units, even if their specific cost per MW(e) installed is generally higher. Therefore, SMRs are likely to be easier to finance by private investors and/or in countries with limited capital availability. It remains that competitiveness with alternatives, including fossil-fuelled power plants, renewable energy sources and large nuclear units, will be a prerequisite for the deployment of SMRs. In order to compensate for the lack of economy of scale, designs should place emphasis on simplification and modularity allowing for fabrication in series of most elements of the plants. Shortening of construction time could be a key factor in reducing the total capital cost and thereby enhancing competitiveness.

SMRs are adapted to decentralised energy demand and their deployment may be feasible in various market conditions where large nuclear units would have difficulties to compete. Generally, in order to ensure grid stability, the size of the largest unit on a network should not exceed 10% of the total interconnected capacity. The trend to industrialisation and urbanisation in developing countries increases the demand for electricity in regions where grids are rather small. SMRs are well adapted to those circumstances where the introduction of large nuclear units would not be possible.

The increasing awareness of environmental issues and more broadly the sustainability goals, including long-term security of supply and protection of people and eco-systems, are giving a stronger weight to non-economic criteria in energy policy making. Explicitly integrating the concept of sustainable development in energy policies is calling for strategies that preserve natural resources and the environment, reduce regional disparities and give equal opportunities to present and future generations world-wide. From this point of view, nuclear energy, including SMRs, may be viewed as a key option to implement in sustainable energy supply mixes because it is a carbon-free energy source that relies on plentiful natural resources, uranium and thorium, that have no other significant commercial use. On the other hand, concerns raised by long-lived radioactive waste disposal and low probability/high consequence nuclear accidents are social and political hurdles that need to be overcome in order to secure a future role for nuclear energy and possibilities for the deployment of SMRs.

SMRs can be used to supply heat and electricity, or heat only, for industrial applications and district heating. Their advantages over large nuclear units for this type of application, besides the better adaptation of their size to the demand, are linked with safety characteristics allowing them to be sited in densely populated areas. Although only a few nuclear units in operation in the world are used for district and/or process heat generation, they demonstrate that other applications of nuclear energy are already possible. The use of SMRs for seawater desalination deserves specific attention in the light of its importance in a number of countries where potable water shortages are already experienced or expected to occur in the near future. Nevertheless, finding adequate sites and gaining public acceptance may remain difficult as well as reaching competitiveness with alternative options. Finally, nuclear energy could play a significant role for large-scale production of hydrogen, if and when hydrogen becomes a major secondary energy carrier.

The latest status of many SMR designs was presented. Twenty-six new and innovative SMR designs (12 water cooled reactors, 5 gas cooled reactors and 9 liquid metal cooled reactors) were presented in some detail at the technical sessions. Among the LWRs the following was the country breakdown: Russian Federation — UNITHERM, VK-300, RUTA-55, KLT-40, ABV-6; USA — IRIS; Japan — PSRD and SSBWR; Argentina — CAREM; ROK — SMART; India — AHWR; and Canada — NG CANDU. With regard to gas-cooled reactors, PBMR (South Africa), HTTR (Japan), HTR-10 (China), GT-MHR (USA-Russian Federation-France-Japan) and a direct cycle carbon dioxide cooled fast reactor from Japan were presented. The country breakdown for the liquid metal cooled reactors was the following: USA — ENHS; Japan — MPFR, LSPR, MDP and 4S; Russian Federation — MBRU-1.5, BMRU - 12, BMN-170 and SVBR-75.

Key features of SMRs include simplification and streamlining of designs as well as emphasis placed on safety features avoiding off-site impacts in case of accident. Such characteristics should facilitate their acceptability by local communities.

Several countries and groups are working on innovative reactor technology development. Particularly two international groups–the US-initiated Generation IV International Forum (GIF) and the IAEA-initiated International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) — are currently working on innovative reactors for the future. The time frame of interest to the GIF is two or three decades from now, and their interest is development of suitable technology (reliable and safe, sustainable, and economic) and the R&D effort needed to accomplish the goals. INPRO is mainly focusing on developing user's requirements for nuclear power for the long term–fifty years time frame. The INPRO developed criteria are expected to be used by individual countries to assess their situation with respect to nuclear power introduction or expansion.

The subject of non-proliferation was discussed in some detail. In spite of the demonstrated effectiveness of the international safeguards regime, the risk of proliferation of nuclear weapons remains a social and political concern deserving to be addressed by governments and the nuclear industry. A significant deployment of SMRs would lead to building a large number of reactors in many different countries and sites. Therefore, gaining social acceptance will require specific efforts of designers to enhance the proliferation resistance characteristics of SMRs. It was stressed that we must remain more vigilant and the suppliers, verifiers and buyers must assure safeguarding of nuclear materials. It was also said that it would certainly be desirable to have new, more proliferation-resistant technology, but given the existing technology, we surely have the international, scientific and regulatory mechanisms to handle the proliferation question and we should move forward as rapidly as possible to build nuclear power where it can meet human and environmental needs.

Effective regulation is a key element. Safety should be enhanced by multi-dimensional approaches including maintenance, operation, and good management practices. Key issues raised and discussed include the need to incorporate safety goals and requirements at the earliest possible stage of concept and design activities and to enhance the dialogue between designers and regulators to avoid delays in the licensing process, in particular for innovative designs. The importance of international co-operation aiming at harmonizing safety goals and requirements was stressed with emphasis on the role of international organizations. For example, if a new design could be licensed for use in one country based on internationally accepted safety goals and requirements, then it could reduce elaborate, repetitive licensing efforts in other countries; it would help implementing nuclear power projects more economically.

A common theme that emerged from the seminar was the importance of establishing the necessary infrastructure (including qualified manpower) to support nuclear power introduction and to optimize local participation. Infrastructure and lack of finance were considered significant constraints to the introduction of nuclear power in developing countries.

A major interest of the participants at the Cairo seminar was cost-effective and stable electricity generation for normal use and not for remote, specialized operation. The participants were mostly interested in near term nuclear plants; the time horizon was certainly within 2020 if not less, and hence the interest was not toward what R&D efforts are needed but rather what could be available in the near future and what needs to be done to build nuclear plants. The time frame for the availability of commercial SMRs is very important as most developing countries could not wait for another two or three decades to increase their installed electricity generation capacities. A combination of technical and managerial improvement is the way to go. The importance of long-term continuity of nuclear energy policy of a country was also emphasized.

Energy markets of the 21st century will be challenging for all technologies and emerging options such as SMRs may have difficulties entering the commercial phase in the light of the emphasis placed by investors on short-term benefits. However, governments of countries wishing to keep the nuclear option in the framework of sustainable energy mixes for the future may consider supporting further research and development on SMR concepts.

There was a consensus on one point: everyone agreed that a new SMR must be demonstrated first, preferably in the country of origin, before another country will buy one. No country wanted to be “a guinea-pig” for a new design.

OPENING ADDRESSES

OPENING ADDRESS
INNOVATION, ADAPTABILITY, AND COLLABORATION: KEYS TO SUCCESS FOR SMALL AND MEDIUM SIZED REACTORS

M. ElBaradei
Director General
International Atomic Energy Agency

Let me begin by thanking the Government of Egypt for hosting this seminar on The Status and Prospects of Small and Medium Sized Reactors. With the advent of telecommunication and the global marketplace, the world has become a much smaller place, and the demand for a higher standard of living is increasing everywhere — yet an estimated two billion people still lack access to electricity. Dramatic increases in electricity demand are expected over the next several decades — with the growth rate in developing countries expected to be three times faster than in industrialized countries. The World Energy Council has concluded that to meet this growth — inevitable for economic and social development — a total reliance on fossil fuels and large hydroelectric facilities is not sustainable, and an expanded future role of nuclear power must be considered. In this context, it may be useful to provide you with a brief overview of key issues that will influence the future of nuclear power, with a particular focus on the role of small and medium sized reactors, the subject of this seminar.

THE CURRENT PICTURE

Nuclear power currently provides about 16% of global electricity, with about 83% of nuclear capacity concentrated in industrialized countries. Around the world, over the past decade, the average availability of nuclear power plants has increased sharply, from 72.5% to 80.5%, due to improved efficiency — the effective equivalent of commissioning 28 new 1000 megawatt units at relatively minimal cost — making existing nuclear power plants economically more effective competitors with other energy sources. This increased efficiency has occurred simultaneously with extensive safety upgrades and improved overall safety performance.

But the overall picture is mixed. Public sentiment against nuclear power remains strong in some countries. Concerns about safety and waste management have led some countries to enact national policy restrictions against the use of nuclear power. Only in East and South Asia are there clear plans for expansion of nuclear power, particularly in China, India, Japan and the Republic of Korea.

On the other hand, the global effort to reduce greenhouse gas emissions and the increase in gas and oil prices has stimulated a renewed consideration of nuclear energy. In this context, a number of leaders have begun to speak out in favour of the nuclear alternative. The Secretary General of the OECD, Donald Johnston, said late last year, “Having examined the best evidence available to me, I have concluded that, if we are to hand to future generations a planet that will meet their needs as we have met ours, it can only be done by incorporating the nuclear energy option.” In January, the Vice-President of the European Commission, Mrs. Loyala de Palacio, stated, “the nuclear option should be examined in relation to its contribution to our prime concerns of security of supply and reduction in CO_2 emissions.” And just this month, U.S. President George W. Bush, in unveiling a new national energy policy, voiced strong support for reconsideration of investment in nuclear power, and speculated that the number of nuclear plants in the U.S. could double in the coming decades.

In my view, the future of nuclear power may depend upon success in meeting four basic challenges:

- The first challenge will be to develop clear national and international strategies for the disposal of high level nuclear and radioactive waste. Final repositories for low level waste have been licensed and are already operational in many countries. High level waste, however, is more controversial. While experts generally believe geological disposal to be safe, technically feasible and environmentally responsible, the public at large remains sceptical, and the volume of waste continues to build. This dichotomy will only be resolved by demonstrating the feasibility of siting, constructing and operating geological respositories. Some ground for optimism exists: the proposed U.S. facility at Yucca Mountain could receive approval later this year as the site for a geological repository, and just this month the Finnish Parliament, by an overwhelming vote of 159 to 3, ratified the decision to construct a deep disposal facility for spent fuel at Olkiluoto.
- The second challenge is to remain vigilant in ensuring the continued safety of operations at nuclear facilities. While safety is a national responsibility, international co-operation on safety related matters is indispensable. The international safety regime consists of three major components: (1) international conventions that prescribe basic safety norms; (2) a body of detailed safety standards; and (3) mechanisms for applying these standards — including peer reviews and other Agency led safety review services that have proven extremely useful in validating safety performance and recommending safety improvements.
- The third challenge involves outreach to civil society — engaging the public and decision makers in a fair evaluation of the relative merits of the different energy options. Improving public understanding of radiation and nuclear issues is essential — creating 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 applications. In the same context, the public must be given credible assurance that nuclear technology and materials will be used exclusively for peaceful purposes, with a strong, adequately financed, and universally supported international verification system.

KEYS TO INNOVATION

I would like to take a more detailed look at the fourth challenge, which entails the development of new, innovative reactor and fuel cycle technologies. To be successful, these new technologies should incorporate inherent safety features, proliferation resistant characteristics, and reduced generation of waste. They must also be capable of generating electricity at competitive prices while satisfying both regulators and investors. On the technical side, this implies a greater reliance on passive safety features, as well as design features that will allow reduced construction times and lower operating costs. But the innovation must be more than purely technical; the new design aspects must be complemented by a re-evaluation of technology policy issues. A high level of confidence must be achieved in the reliability of construction schedules, licensing review procedures, liability issues, and other factors that affect the cost of design, construction, startup, operation and maintenance.

Small and medium sized reactors, within a power output of less than 700 MW(e), are receiving increased consideration in this effort to meet changing market requirements. Smaller plants allow a more incremental investment, which can be used to hedge against demand uncertainty. They are more suitable for standardization and prefabrication, which in turn encourages enhanced quality control and stimulates rapid development of expertise and shorter construction schedules. They provide a better match to grid capacity in developing countries. And they are more easily adapted to a broad range of industrial settings and applications, such as district heating, heavy oil recovery, or the production of hydrogen and other chemical fuels.

Sea water desalination is an application for which smaller reactors hold a particular advantage. Nuclear powered desalination is a proven technology: Japan has accumulated over 100 reactor years of desalination experience at nine reactors — although not for commercial use — and the BN-350 plant in Kazakhstan was used for many years for both electricity production and sea water desalination. But many countries most in need of freshwater also have limited industrial infrastructures and electrical grids, which has inhibited the feasibility of accommodating large scale reactor facilities. In such cases, a nuclear desalination plant becomes much more feasible if it can be built in the range of 50 – 150 MW(e).

Here in Egypt, the IAEA has been supporting the Nuclear Power Plant Authority (NPPA) in considering the feasibility of a dual purpose nuclear power plant for electricity and desalination. This co-operative project is examining the range of energy and water needs, candidate reactor types and desalination processes, local participation capabilities, cost comparisons and financing options. This is a sound strategy. Even for countries with strong oil and gas producing capacities, research and development in innovative energy technologies — including nuclear technologies — is becoming a prudent investment, as a strategy for improving the long term availability, quality and competitiveness of their oil and gas reserves. This investment produces its best return through partnerships and alliances.

Collaboration has become a key feature of the global effort to develop new reactor and fuel cycle designs. Some 20 to 30 innovative designs are under development, with all of the principal reactor concepts — water, liquid metal, or gas cooled — being addressed in one or more of these projects. One major initiative is the International Project on Innovative Nuclear Reactors and Fuel Cycle Concepts (INPRO), an international umbrella project under the auspices of the IAEA. The key to the success of this effort is co-operation and collaboration — promoting technical information exchange, sharing safety and non-proliferation insights, leveraging research dollars, and — perhaps most important — enhancing our understanding of user needs and requirements. Clearly, at some point in the development of a given technology, collaboration gives way to commercial competition; however, even after these technologies become competitive, collaboration will continue to be beneficial for new designs with enhanced features to reduce costs, improve safety and promote non-proliferation.

CONCLUSION

Clearly, we live in an era in which our society faces many difficult economic, environmental and social issues associated with sustainable development and energy demand. Against that backdrop, nuclear power is a mature technology that deserves careful consideration as a contributor to solving some of these issues. The development of innovative small and medium sized reactors will play a key role in helping to match state-of-the-art technology to user needs. I hope this seminar will provide a fruitful exchange of information and ideas as a step towards further progress.

OPENING REMARKS
AND ADDRESS ON BEHALF OF
THE PRIME MINISTER

A.F. El Saidi
Minister of Electricity & Energy
Egypt

Dr. Mohamed ElBaradei, Director General of the International Atomic Energy Agency IAEA, Mr. Luis E. Echivarri, Director-General of the OECD Nuclear Energy Agency, Ambassador John B. Ritch the Third, Director-General of the World NuclearAssociation, Distinguished Colleagues, Guests, and Participants, Ladies and Gentlemen,

We are very happy to have you all with us. It is a great pleasure to join with my colleagues this morning in this International Seminar on the Status and Prospects for Small and Medium Sized Reactors.

First, please allow me to recognize the sponsors of this program, and the organizers and planners of this event. As a former member of the International Atomic Energy Agency, I have an abiding appreciation for the value of services the Agency provides such as the current seminar we are about to commence. I would like to thank the Agency for giving Egypt the opportunity to host this leading seminar, representing the first of its kind in the twenty first century and the first in the region. Also, I would like to recognize the OECD Nuclear Energy Agency and the World Nuclear Association together with the Egyptian Nuclear Power Plants Authority, who have joined in putting together this Seminar.

Our Prime Minister, Dr. Atef Ebeid, had it planned to address this meeting but he was asked by President Mubarak to represent him in the G-15 meeting in Jakarta. He asked me to convey his apologies for not being able to share with you the inauguration of the Seminar and wishes you, on behalf of the Egyptian Government, a successful and fruitful meeting as well as a pleasant stay in Egypt. The Prime Minister also gave me the honor to read his address to you. Please allow me to do so.

OPENING REMARKS

A. Ebeid
Prime Minister of Egypt

First, allow me to welcome you all to Egypt and to thank you for joining us this morning. It is a special privilege for Egypt to be selected to host this prestigious assembly that gathers this eminent group of specialists and officials from all over the world.

It is a great pleasure and personal satisfaction to see today that the International Atomic Energy Agency represented by its Director General, Dr. Mohamed ElBaradei, the OECD Nuclear Energy Agency represented by its Director General, Mr. Luis EchAvarri, and the World Nuclear Association (formerly, the Uranium Institute) represented by its Director General, Ambassador John Ritch the Third, all cooperating with us in organizing this Seminar. Gentlemen, thank you for your proactive participation and your commitment to be part of today's event.

This event is a continuation of ongoing activities by the concerned groups all over the world in seeking solutions to the energy needs of our world and especially to the developing countries which Egypt is part of.

Energy is one of the means to foster human development; the whole concept of energy issues should be viewed and tackled accordingly. There are disparities among the nations in which energy plays a key role. There are the rich and the poor, the industrial, technologically advanced, well-developed countries and those that are slowly industrializing and developing.

Nature has bestowed on mankind primary energy in various forms. Some forms of energy are readily useable, but most others are in forms that require further processing. In addition, some are quickly depleting, some are slowly depleting and some are renewable. Until recently, the world has concentrated on the use of depletable forms of primary energy, mainly fossil fuels. However, technological innovations and environmental considerations have pushed other forms into use.

Ladies and Gentlemen, as you know, there are many factors that affect the energy demand. Among these factors are the following major ones:

FIRST, POPULATION GROWTH

One of the major forces driving the increase in energy demand is population growth. Currently, about 6 billion people inhabit the world, 30% of which representing 2 billion people are without access to commercial energy. By the year 2020, our world population is expected to reach 8 billion, and 10 billion by 2050. It is estimated that 90% of this population explosion will occur in developing countries. This momentous growth in population, together with the projected technological and economical development, place enormous pressure on energy demands. If we want to alleviate poverty, we must work together to build the climate for investment that will create jobs, provide sustainable growth, and help develop and deploy advanced energy technologies.

SECOND, ECONOMIC DEVELOPMENT

Among other major factors influencing energy demand is, by all means, the economic growth. Comprehensive development strategies in many developing countries, and particularly here in

Egypt, opened the door for a spectrum of investment opportunities in different fields of the economy. A more marketoriented economy naturally leads to increased foreign investment and private sector participation driving respective economies towards stability and growth. Though, and in spite of the current apparent slowdown in some parts of the world, the energy sectors and the energy resources will remain major contributors to economic recovery and growth.

THIRD, TECHNOLOGICAL ADVANCEMENT

Technological change and development is at the heart of productivity increases and economic growth. Improvements in performance including energy efficiency and cost reductions have a profound effect on the efficiency of the economy, and eventually, on the competitiveness in the global market.

FORTH, ENVIRONMENTAL RELATED ASPECTS

Presently, there is a growing concern among larger numbers of people in the world about environmental issues. For slowly industrializing developing countries, air pollution from industries has now started to join traditional indoor air pollution as a major challenge to be met. At the same time, environmental objectives in the industrialized countries have shifted more towards very longterm and global issues, most prominently, climate change and global warming.

Local environmental problems should have first priority particularly in developing countries. The most effective solutions will be those that are most comprehensive, relying on energy conservation based on a mix of efficiency improvements, and a shift to cleaner fuels. These solutions will yield triple dividends: conservation of resources, lower overall energy systems costs, and lower emissions.

FIFTH, ENERGY AND WATER RESOURCES

As you know, oil and natural gas availability is expected to decrease during the second half of this century. There will be an increase in the number of countries depending on fossil fuel imports, particularly oil and natural gases. Supply lines will be longer. These shifts will create a new feeling of uncertainty over supplies and may be reflected in upward pressures on prices. It is imperative that we need to consider the employment of all other viable options in meeting the increasing demand in Energy including the nuclear option whenever appropriate.

In this regard, it is worth referring to the conclusions and recommendations presented almost three years ago during the 17 th World Energy Council in Houston, Texas, which stated that nuclear power is an important element in the energy mix in developing countries contributing to non-greenhouse emitting energy supply. However, each country has to take into consideration its prevailing conditions and constraints when constructing its long-term policies and plans for energy supply.

On the other hand, with regard to the continuous decrease in the share of potable water per capita in some regions of the world and in particular our region, interest in technological means for producing potable water, such as desalination by conventional and nuclear means, have continuously been growing worldwide. The assurance of reliable potable water supply is imperative to our plans for sustainable development. The technical and economic viability of nuclear desalination will be key factors in bringing it to the scene as a suitable option that will help countries in meeting the expanding regional needs for potable water.

Ladies and Gentlemen,World experience in the past decades showed that the employment of nuclear power to satisfy part of the Energy demand as well as water requirement dictated the full adherence to main factors to support their successful development and use. These factors are:

1. Long Term Commitment

Long term commitment to a nuclear program and the associated upgrade in the country infrastructure and its technological base are becoming a pre-requisite for the successful implementation and use of the nuclear power especially in the developing countries.

Technology transfer and development of the know-how infrastructure also have a crucial role in enhancing the economic perspective for this type of energy technology and that will eventually pass through to the consumer. The introduction and transfer of technical know-how will require further reliable skilled manpower and adequate training programs to augment it. Therefore, planning for buying one or two nuclear plants without engagement in a comprehensive program will be costly and impractical and would bring more problems instead of solving energy demand.

2. Commitment to Safety

Any nuclear program must aim at ensuring high levels of safety and reliability at all stages of development. Key safety features include the establishment of a strong independent regulatory agency and a nuclear regulatory and licensing framework to conform to international nuclear safety standards. These requirements need to be supported by human and financial resources that should be taken in consideration.

Ladies and Gentlemen, the implementation of such program would not only necessitate the above said requirements, but normally encounters many challenges, most important of which are:

1. Capital Intensive Requirements

It clearly remains to be seen that developing countries are faced with a number of challenges that need to be fulfilled for the successful development and implementation of nuclear power. One of these challenges that are, and will remain, the center of concern is the high capital cost associated with the development of these nuclear power plants. This high capital investment relatively places significant burden on developing nations hindering their efforts towards a sound economy.

Nonetheless, localization and optimal domestic participation using available manpower, industrial capabilities and materials are vital for economic viability of all kinds of capital-intensive projects. Such initiative would be dramatically enhanced -as said earlier- through the country's commitment to a long-term plan that will include a series of modular plants together with supporting industries and services. This can be witnessed through Egypt's localization efforts in fossil fuel power plants that have proven to be successful in bringing costs down.

2. Radioactive Waste

Also, radioactive waste, representing the back end of the fuel cycle, is another issue that needs much attention for safe and reliable operation of nuclear power plants. Viable waste management arrangements and further technical progress in this area will need to be pursued for safe and feasible disposal. Evidently, this will have its share in reinforcing public acceptance and understanding, which is necessary to any decision-making.

3. Public Acceptances

As you know, the public understanding of nuclear energy will remain an intricate issue that needs much work for it to develop and mature. Acceptance or opposition of nuclear energy applications will ultimately depend on the perception of people and how this perception will evolve with time. The public need to understand all sides of nuclear plants and must be generally satisfied that these plants are essential, economic and safe. This will definitely pose a challenge on the availability of communication channels and skills needed to effectively persuade and properly present this message to the public.

Ladies and Gentlemen, once again, our developing nations, Egypt included, are in a period of economic evolution determined to close the gap between wealth and poverty. This will be achieved through your cultivated efforts to present evolutionary and innovative energy technologies that are cost effective and environmentally friendly. Together with the introduction of appropriate economical and financial scenarios, energy costs can be relatively reduced and that is what we all aim at for raising living standards to acceptable levels.

Your efforts in discussing ways to address the basic requirements and meeting the challenges related to nuclear power development, as I have outlined, will be the culmination of fulfilling the energy demands of the citizens of the world. What we create is what our children and grandchildren will inherit, so let us be responsible stewards for the decisions we make and the leadership we show today.

With these remarks, I hereby open the International Seminar on Status and Prospects for Small and Medium Sized Reactors.

OPENING ADDRESS

L.E. Echávarri
Director-General, OECD/NEA

Mr. Minister, Mr. Director General, Ladies and Gentlemen, Dear Colleagues, it is an honour for me to address you this morning on behalf of the Nuclear Energy Agency of the OECD at the opening of the Seminar on Status and Trends for Small and Medium Sized Reactors, jointly organised by the International Atomic Energy Agency, the World Nuclear Association and my Agency, and hosted by the Ministry of Electricity of Egypt.

First of all, I would like to express my sincere appreciation to our Host Country, Egypt, which has made it possible for us to meet in the magnificent city of Cairo and provided the framework in which we can have a fruitful exchange of information during the coming days.

I must admit that, when I learnt that the Seminar will be held in Cairo, I was concerned about keeping our audience in the meeting room. Competing with the attractions of a lively city full of memories of the past and where the weather is more adapted to relaxing vacation than to work is undoubtedly a challenge.

However, after looking at the final programme of the Seminar, I was convinced that the variety and quality of the speakers, as well as the originality and interest of their papers, provide a strong incentive for us all to postpone visits to Cairo's museums, mosques and pyramids.

Investigating a broad range of issues related to the prospects of small and medium sized reactors is very timely. Worldwide, policy and economic factors, as well as scientific and technical progress, are leading decision makers to revisit the nuclear option. I will not elaborate now on those factors that will be discussed extensively later this week, but I'd like to stress the increasing importance, especially in the OECD context but progressively in all countries, of sustainable development goals and their implementation in national energy policies.

The interest of reactor designers and policy makers in small and medium sized reactors is not new. My Agency has carried out several studies on SMRs and the IAEA has continuously been looking at status and trends in this field. There have been, however, a number of new developments from the technical and policy viewpoints that deserve special attention. Again, I will refrain from an exhaustive review of these since they will be covered by distinguished speakers in the technical sessions of the Seminar. Let me just mention the relevance of assessing SMRs in the framework of new international initiatives on innovative reactor designs such as Generation IV International Forum and INPRO.

My Agency is very pleased and proud to be a co-organiser of this Seminar. I am sure that the meeting will offer multiple opportunities for exchanges of information and lively discussions on various aspects of SMR development. Moreover, our findings and conclusions will undoubtedly be of relevance for policy makers in governmental bodies and the industry and for the future of nuclear energy.

I hope that we will all enjoy fruitful discussions, open and friendly exchanges of information with colleagues from other countries and, last but not least, once the Seminar is over, that you will have opportunities to discover or re-visit the wonders of Cairo and Egypt.

OPENING REMARKS

J.B. Ritch III
Director General, World Nuclear Association

Ladies and gentlemen, the World Nuclear Association Is proud to stand with the IAEA and the OECD's Nuclear Energy Agency as a co-sponsor of this important event, which looks to the future and to how one of the great discoveries in human history can be employed ever more expansively to serve human development and human needs.

It is particularly appropriate that a seminar on the future of nuclear power be held in Cairo - for two reasons.

The first is personal. Egypt has given to the world, in the person of Mohamed Elbaradei, a distinguished leader in international nuclear affhirs. It is rare indeed to find in one individual someone with a keen grasp of international law, geo-politics, and global development - who can meld that understanding with a sensitive and complex subject like nuclear energy in a way that promotes enlightenment and progress 'in the human community worldwide. Egypt can be proud that, as Director General of the International Atomic Energy, Mohamed Elbaradei is doing precisely that.

Second, Egyptians as a nation and people stand at the bridge-point between the many separations in our world - between past and future, North and South, East and West, poor and wealthy, developed and developing - and, in the realm of nuclear affairs, between those nations with nuclear energy and those without. With a fine scientific commuru'ty that is well advanced in understanding nuclear technology, Egypt today is actively examining the use of nuclear energy for both desalination and electric power. I hope and trust that this conference will support that exploration - for Egypt and for many others.

For me, this conference represents a special opportunity. A few months ago, I completed seven years in Vienna as U.S. representative to the IAEA and other UN agencies there. Before I left, Mohamed Elbaradei summoned me to his office and instructed me that he wanted me to go out into the world and to try to make myself useful. I obeyed Mohamed's order by accepting ajob as head of the Uranium Institute in London and launching an effort to transform it into a pnivate-sector counterpart to the IAEA - into a world organization that would provide, on the non-governmental side of life, a truly global nuclear forum and a commercial and technical meeting place for those everywhere who are engaged in the field of nuclear power.

When I arrived in January, the Uranium Institute was, to a substantial extent, fulfilling that kind of function - but with a membership of comparuies and other organisations located almost exclusively in the OECD world. My goal, indeed my highest priority, is to expand our membership so as to include companies and other nuclear organizations form every country in the world that is either producing nuclear power or considering doing so. We signaled the onset of thisproject earlier this month, when our full membership voted unanimously to change our name to World Nuclear Association.

I am pleased to say that our very first new member after my arrival was theAtomic Energy Organisation of Iran. But that was just a beginning. I regardvirtually every organisation and agency represented at this seminar as a potentialcandidate for membership in the World

Nuclear Association, and I intend to use every opportunity in the days ahead to pursue and examine that possibility.

Our world today is immersed in problems and opportunities, and nucleartechnology stands in the very center of it all – as a technology that can meet and solve a myriad of human problems if we grasp the opportunity to exploit it wisely and well. If that is to occur, we need transnational communication, commercial cooperation, technical and personal exchange – and it is our aim that the World Nuclear Association act a global hub for these much-needed connections.

Among governments, the IAEA is the nucleus of the nuclear world. The WorldNuclear Association will be seeking to perform the same role among companies, research institutes, professional societies, and government agencies. I am pleased to say that Hans Blix, Dr. Elbaradei's predecessor at the IAEA, has joined our cause by becoming the World Nuclear Association's honorary chairman.

In the days ahead, with my colleague Adrian Collings, I hope to talk to as many of you as possible about WNA membership, and I invite you to approach us. We want to talk about the business of partnership – about how we can help each other. I look forward with all of you to a successful seminar this week, and I look forward too to welcoming many of the organizations you represent as new members of the World Nuclear Association.

KEYNOTE ADDRESSES

(Session 1)

Chairpersons

E. El Sharkawi
Egypt

J. Kupitz
IAEA

GLOBAL ENERGY SUPPLY AND DEMAND AND THE POTENTIAL ROLE OF NUCLEAR POWER

J. MURRAY
World Energy Council

Abstract

The paper gives a global perspective of the energy share, mix and demand in different regions of the world in light of the population growth and economic development expected in the coming decades. It is suggested that all energy options be kept open. The important potential role of nuclear power is discussed.

1. INTRODUCTION

I must congratulate the International Atomic Energy Agency on the timing of this conference! The publication of the Bush Administration's energy plan the week before last has certainly shifted the centre of gravity of the debate over nuclear energy. The World Energy Council welcomes this, as we have for some time been saying that nuclear energy must be considered as one of the four broad options available to us for achieving a sustainable energy future. While it is clear that the new US energy plan does not herald the end of the debate, it is at least bringing into the open issues which have been ignored for too long. Indeed, in much of the world and, in particular, the industrialised world, there has been almost a disconnect between public thinking and reality. There appears to be a widespread assumption that we are in a transition phase from twentieth century energy sources like oil, coal, gas, nuclear and large hydro to the twenty first century's renewable sources, based on sun, wind, wave and micro hydro and that supply and environmental problems will thus be solved. The facts make this far from clear. Let us try to establish some of the facts.

2. PRESENT DAY SUPPLY AND DEMAND

As we enter the twenty first century, global primary supply is approximately one third oil, one quarter coal, one fifth gas, one tenth the traditional sources of wood, charcoal, dung and crop residue and around 8% and 7% hydro and nuclear respectively. The so-called new renewable energy sources supply only around 1% whereas if we aggregate the fossil fuels, they account for nearly 80% of our current energy supply.

There are considerable differences in the shares of these energy sources in the different regions of the world. Not surprisingly, the share of woodfuel is highest in the developing world. While woodfuels represent about 7% of primary energy consumption (about the same as nuclear), 76% of woodfuels are used in developing countries, where their share is 15%. In some individual countries the share exceeds 80%. By contrast, the lion's share of oil (63%) is consumed in the industrialised countries and in the skies above them. The industrialised countries' share of gas consumption is less, at 56%, thanks to the relatively high use of gas in the CIS. For coal, the industrialised country share falls still further to 46%, largely because of China, which is the world's biggest single consumer of coal. China uses some 20% more than the USA, the second biggest user. Of course, even here, the industrialised countries are the dominant consumers in per capita terms.

It will probably not come as any surprise to you to learn that nuclear energy is, to an even greater extent, the preserve of the industrialised world. Around 83% of nuclear electricity is produced in a dozen industrialised countries. The remaining production is shared between Eastern Europe and the CIS (11%) and developing countries including China (6%).

Not only are there these differences in energy mix, there are also major discrepancies in total energy consumption. The poorest two billion people in the world use only 0.2 toe of energy per capita annually, whereas the billion richest people use 5 toe, or nearly 25 times as much. To take another measure of inequality, the richest 20% use 75% of all electricity, while the poorest 20% use less than 3%.

There are also very considerable differences in the way in which energy is used in different areas. In the industrialised countries, only some 20% of energy is used for residential purposes. The biggest share, 34%, is used for transport, 31% for industry and the balance for the commercial sector, agriculture and other miscellaneous uses. Compare this with the energy use pattern of a rural inhabitant of a poor country. An overwhelming 85% is used for residential purposes, most of it for cooking. "Cooking" barely registers in the rich world's energy consumption.

3. DETERMINANTS OF ENERGY DEMAND

So this is our starting point, but to look at future energy supply and demand we need to look behind this static picture at trends in the principal determinants of energy demand. In 2000, the World Energy Council published *Energy for Tomorrow's World; Acting Now!.* In it we looked at the underlying assumptions of our 1993 report *Energy for Tomorrow's World*. While some had stood the test of time well, in some important areas revisions were needed based on data from the past decade.

3.1. Population growth

Since 1993 the deceleration of birth rates has become much more clear. In the regions with the fastest growing populations today (including Africa and the Middle East) the number of children per woman is dropping quickly. For instance, this number was around eight in North Africa in the late 1960s, but is now less than four. By 2020 it is now expected to be around two.

This opens up the possibility that the world's population will peak earlier than previously expected and at a lower level. Recent UN data now show that population growth has actually been decelerating since 1990, the year of maximum annual growth. Present trends suggest that total population may not exceed 8 billion people around 2050 and may start to decline shortly thereafter. While this still represents a considerable increase on today's population of 6 billion, it is significantly less than the doubling once seen as almost inevitable.

It is important to note, however, that virtually all of this growth will occur in developing countries. Industrialised country populations have peaked or will do so shortly. Moreover, the greater part of the population increase will be urban. The proportion of people living in rural areas has already peaked and will decline in future. Another indication is that today there are five mega cities of more than 15 million habitants, but in 20 years there will be 15, all located in developing countries. These trends also have important implications for energy supply to which I shall return.

In energy terms, already we have nearly 2 billion people without access to a regular electricity supply and approximately half of the world's population still depends on woodfuel for cooking. Even with these lower population projections, the challenge to achieve access to modern energy for all is clearly substantial.

3.2. Economic development

Economic growth was lower than we expected over the period. In the decade from 1989 to 1998, growth in official GDP figures adapted for purchasing power parity was 2.8% per annum. While there were some arguably anomalous factors in the 1990s (the slow take of the former centrally planned economies and the Asian financial crisis), this in fact follows a consistent pattern over a number of decades. World economic growth was above 5% in the 1960s, about 4% in the 1970s and a little over 3% in the 1980s. Only time will tell whether this trend-line can be shifted upwards by accelerated growth in the developing countries and information technology led productivity increases more generally. The first of these would almost certainly increase energy demand. The net effects of the e-economy on energy demand are still less clear, but it will certainly place a much greater premium on reliable electricity supply and there is some evidence that it will also push up electricity demand.

3.3. Energy intensity

While good progress was made in the decade in reducing the amount of energy needed to produce a unit of GDP (energy intensity), there were several one-off factors which helped to achieve that, in particular, the restructuring of the previously very energy-inefficient centrally planned economies. On balance, the World Energy Council concluded it had overestimated the scope for breakthroughs which could drive a much greater decrease in energy intensity. If we also take into account the fact that the decrease in carbon intensity of our energy mix which has been apparent in the 1970s and 1980s appears to be levelling out, then far from witnessing a transition to benign, low-emissions renewable energies, we appear to be heading in the other direction!

4. THE POLICY FRAMEWORK

In *Energy for Tomorrow's World; Acting Now!*, two elements of the policy framework, in particular, were seen as having increased in importance in the 1990s.

4.1. Energy market liberalisation

The move towards energy market liberalisation gained momentum over the decade. By the time the World Energy Council's report on *Benefits and Deficiencies of Energy Market Liberalisation* was published in 1998, 16 countries had already liberalised their electricity markets to some extent, but 61 planned to follow suite, while 5 countries had liberalised their gas markets and another 12 planned to do so. There is much to be learnt from this experience, California being the most spectacular example of badly designed market reform. The World Energy Council's most recent report on the subject, *Electricity Market Design and Creation in Asia Pacific*, was commissioned by our Asia Pacific members and seeks to explore how competitive elements and regulation can be blended to simplify markets and reduce the cost of reform.

4.2. Environmental policy

The environmental agenda moved centre stage in the 1990s, particularly with respect to greenhouse gas emissions and the possibility, or even probability, of climate change. The Third Assessment Report of the Inter-Governmental Panel on Climate Change (February 2001) presents the strongest evidence yet that climate change is occurring (for example, temperatures have risen in the lowest 8 km of the atmosphere, snow and ice cover have decreased, and the seal level has risen between 0.1 and 0.2 metres in the century). The report also finds that concentrations of atmospheric greenhouse gases have continued to increase as a result of human activities and that there is new and stronger evidence that most of the warming observed can be attributed to human activities.

If the science of global warming has become a little firmer, how the world's nations will decide to respond to it has not. In March, the US withdrew from the Kyoto Protocol, throwing into further doubt a mechanism which already looked problematic. There are thus still no binding international commitments to reduce emissions. It should be remembered, however, that the parent treaty – the UN International Framework Convention on Climate Change of 1994 – still stands and commits the signatories to stabilise concentrations of atmospheric greenhouse gases "at a level that would prevent dangerous anthropogenic interference with the climate system". Thus, with or without the Kyoto Protocol, the expanded demand for energy services noted above will need to be met in ways that does not add to global greenhouse gas emissions, let alone exacerbate the various forms of localised pollution resulting from today's energy use. How can this be done?

5. OPTIONS FOR A SUSTAINABLE ENERGY FUTURE

Four broad avenues are potentially open to us to supply the energy the world needs sustainably and affordably:

1. ***Renewable energies*** (in particular, wind, solar, wave, geothermal, modern biomass and hydrogen from non-fossil fuel sources) generally have low or no emissions and are potentially well suited for meeting the energy needs of rural populations. They are, however, typically diffuse, intermittent and relatively expensive energy sources. Considerable technological advance in collection and storage technologies is needed, as is their harnessing in hybrid systems with conventional fuels.
2. ***Conservation and energy efficiency*** are not strictly energy sources but represent substantial potential for the same task to be achieved with either less energy or to produce the needed energy with less fuel. Either way emissions can be reduced, provided that demand for the service doesn't expand because it now costs less. Energy pricing policies and public information have a critical role to play.
3. ***Cleaner fossil fuels systems*** enable us to use fossil fuels with less environmental impact. Fossil fuels currently dominate the market, because they are cheap and convenient to use. Moreover, reserves will last for a long time to come. If we can use these fuels cleanly, they can play an immensely valuable role in a sustainable energy future. Greater fuel efficiency has already led to significant emissions reductions for a given energy output. This work needs to continue, but the ultimate prize is carbon sequestration. Technological breakthroughs are needed.
4. ***Nuclear energy*** is one of the few options currently available for bulk electricity supply without greenhouse gas emissions and it is supported by ample uranium resources worldwide. I return to this subject below.

6. KEEPING ALL ENERGY OPTIONS OPEN

There are many uncertainties regarding our energy future, of which the following appear to be some of the most significant:

- ***Will the world become persuaded that climate change is a reality?*** As noted above, the evidence for global warming appears to be growing stronger, but few countries have realistically tackled how their emissions may be reduced. Emissions in the European Union, for example, look set to rise as the nuclear reactors are retired. Little sense of urgency is yet apparent at the level of action.
- ***How competitive will "new renewables" become?*** It has long been argued that costs will decline as larger scale production is achieved. Costs have, indeed, come down, however, they generally remain above those of their fossil competitors. To some extent this question is linked to the previous one – if the world takes climate change (and other energy pollution) more seriously, then fossil fuels will be required to internalise their "externalities", thus helping renewable energy to be more competitive.
- ***Will carbon sequestration be economic?*** The jury is out. In any case, it looks more feasible for stationary power sources (for example, in depleted oil and gas deposits or other geological formations), than for transport, though the development of carbohydrate-based hydrogen fuels could change this.
- ***Will distributed generation kill the grid, as some observers suggest?*** Distributed generation has the potential to revolutionise the electricity industry and we expect it to play a greater role, but whether its economics will be sufficiently strong to make the grid redundant is far less clear.
- ***Will a major transport fuel emerge to substitute for oil?*** Transport is still overwhelmingly dependent on oil, despite the work undertaken for many years to develop electric alternatives. Current hope is focused on fuel cells and hybrid cars.

Against this background, the World Energy Council advocates that all the above avenues must be pursued at this time.

7. THE POTENTIAL ROLE OF NUCLEAR POWER

The Economist magazine last week (19 May 2001) featured a nuclear reactor on its front cover and posed the question "A new dawn for nuclear power?" Is this so?

There are certainly a number of positive factors. The strong backing of the Bush administration should be very helpful in breaking through some bottle necks, in particular in licensing. The California electricity crisis has already had a clear effect in making the public more aware that "electricity doesn't grow on trees". In western USA, opinion that "we should definitely build more nuclear energy plants in the future" had risen from 33% to 52% by the beginning of this year, nationwide it rose from 42% to 51%. Already the value of existing nuclear plants has soared, as utilities seek to get their hands on these cash cows.

In the European Union, the Green Paper on Strategy for the Security of Energy Supply of November 2000 at last faces up to the fact that " the European Union is not in a position to respond to the challenge of climate change and to meet its commitments, notably under the Kyoto Protocol". With respect to nuclear energy, it states, "The nuclear option must be examined in terms of its contribution to security of supply and greenhouse gas emission reductions. Nuclear energy saves Europe around 300 million tonnes of CO_2 emissions per year. This is equivalent to taking 75 million cars off the road."

Further east in Japan, the Japanese Government approved two new reactors in mid-May 2001 to be built in Kaminoseki in southern Japan.

But the sky is not without some clouds. The anti-nuclear movement has been quiescent for some years now, thinking that the dragon had been slayed. It will surely spring back into life if there is a revival of nuclear power. Meanwhile, it has influenced thinking in areas such as the "ethical investment" movement some of which is now mainstream. For example, the UK Stock Exchange recently set up a FTSE4GOOD for "socially responsible investment", where the "environmental screen" is assumed to rule out nuclear power. Even The Economist article was in fact equivocal. Doubtful about whether nuclear reactors can deliver competitiveness in today's liberalised markets, it answers its own question with – "probably not". While the European Commission was publishing its Green Paper cited above, the European Parliament was passing a resolution to prevent nuclear energy being considered under the Clean Development Mechanism of the Kyoto Protocol (Resolution B5-0803, November 2000).

So what can be done to make the "new dawn" more likely?

I am sure I do not need to tell this audience that practical progress is needed in waste management programmes. Hopefully within the next decade the countries with the most advanced waste management programmes will implement their plans in a way that gives greater reassurance to the public.

The highest standards of reactor safety must be maintained, with all the problems this involves of maintaining highly qualified personnel, rigorous quality standards and avoiding complacency.

Furthermore, the industry must remain ever vigilant regarding the diversion of civilian material to military purposes. If nuclear energy is to play a wider role, the public must be confident that this is not adding to the risk of use of nuclear weapons.

Above all, however, nuclear energy must be economically competitive and suitable for use in meeting at least some of the energy needs of developing countries. As noted earlier, virtually all of the 21st century's population growth will occur in developing countries. If nuclear energy cannot play a role there, it is destined to be a sideshow. For this, advances in reactor design will be important.

This month the nuclear industry accumulated 10,000 reactor years of operating experience. This experience can and is being used to improve the large reactors which have become the mainstay of the industrialised world's nuclear programmes. Tomorrow's megacities of the developing world will be major centres of electricity demand within a decade or two. Large reactors will be relevant to meeting this demand.

The experience can also be used, however, to explore new directions, in particular small and medium sized reactors, which could be used in a wider range of circumstances. In this respect, I believe it is important not to let the issue become one of small versus large reactors. In producing a small car, Mercedes Benz has extended the market segments it can serve, it has not sacrificed its market for large cars.

Ideally, the industry would offer a range of product responsive to different segments of the new competitive electricity market and, through drawing on the 10,000 years of operating experience, they would be more flexible, user-friendly and proliferation-resistant. This is the challenge I present to the delegates at this conference!

SMR FEATURES TO MEET SPECIFIC CHALLENGES

V.M. MOUROGOV, P.E. JUHN, J.M. KENDALL
International Atomic Energy Agency

Abstract

In recent years, many IAEA Member States have expressed interest in the development and deployment of small and medium sized reactors (SMRs). The challenges that must be addressed in the successful development of SMRs are discussed, with the fundamental challenge being the achievement of competitive power generation costs. A brief survey of the range of concepts that have been considered for SMRs is provided, including a listing of concepts that were identified in an earlier IAEA study and a discussion of the general categories of SMR concepts. This information is followed by a discussion of general features that can help to meet the challenges.

1. INTRODUCTION

SMRs are of considerable interest to many IAEA Member States, as evidenced by resolutions passed by the IAEA General Conference in recent years. In response to this indication of interest among its Member States, the IAEA has been actively following and supporting the development of SMR concepts. SMRs have been defined by the IAEA in accordance with their net electrical power rating, or equivalent thermal power rating for reactors producing process heat. Based on input from IAEA Member States, small reactors are defined as having rated electrical generation capacity of less than 300 MWe or thermal output of less than 1000 MWt, while medium reactors are from 300 to a maximum of 700 MWe. In general, at a constant busbar power cost, smaller unit sizes would be preferred for a number of reasons (e.g., better matching of system demand growth, smaller unit capital costs, reduced transmission needs). Some power cost premium could therefore be assigned to smaller units if total costs are taken into account. However, the challenge of providing competitive costs at decreasing unit size places practical limits on how small a nuclear unit can be. Since real costs of future nuclear plants will depend on a number of variables not yet known (e.g. design specific regulatory requirements, regulatory environment for construction and operation, component costs under mass production, future local energy prices) the minimum size of an economically competitive unit is not knowable in advance. Thus attaining a competitive position with regard to cost of unit construction and operation is the fundamental challenge for the successful development of SMRs. A wide range of design concepts have been developed in IAEA Member States to address this challenge, with some degree of commonality among the features employed.

2. CHALLENGES

The fundamental challenge for SMRs is to achieve economic competitiveness with other energy options. However, as illustrated by recent global and regional price movements in fossil fuels, economic competitiveness is a moving target, and when considering the deployment of emerging concepts a decade or more into the future for deployment at unspecified locations, considerable uncertainty exists with regard to the definition of a cost target. On the other hand, factors such as use of conceptual design estimates for future prices of major components under mass production in a mature market, or estimates of construction and operation costs associated with yet undetermined licensing requirements, introduce large uncertainties with regard to cost estimates. Thus economic competitiveness cannot be quantitatively established with confidence for any of the SMR concepts. However, it is necessary to address the issue because if they are not economically competitive, SMRs are unlikely to be deployed on any significant scale in the deregulated energy marketplace expected to prevail in most parts of the world in the future.

Given the limitations in the direct evaluation of economic competitiveness noted above, it is advisable to identify and address the underlying challenges that contribute to costs. Some of the primary contributing factors are:

- **Demonstration of Adequate Safety** – The definition of "adequate safety" remains elusive, with resulting uncertainty in the design and licensing process for future reactors. Recent studies concluding that the safety record of existing nuclear power plants compares very favourably to all other generation options have yet to be reflected in requirements for future designs. A generally reasonable requirement that future plants provide a level of safety as good or better than the best existing plants, combined with uncertainties arising from a lack of construction or operating experience with new concepts, can translate in practice to considerably higher standards for future plants, and ultimately to higher costs.
- **Disposition of Spent Fuel** – The continuing barriers to long term disposition of spent fuel from existing reactors in most parts of the world represent a substantial challenge for all future reactors. While reactor and fuel cycle design provisions can reduce the technical difficulty of disposition, much of the problem involves political and institutional considerations that are not amenable to technical solutions.
- **Provision of Adequate Safeguards Against Material Diversion** – The current international safeguards regime has been demonstrated to be effective in preventing material diversion from existing nuclear power plants. However, a more widespread deployment of large numbers of SMRs could present new challenges that may be lessened by design provisions.
- **Infrastructure Requirements** – Existing nuclear power plants require a considerable investment in the development of indigenous nuclear institutions to support licensing review and enforcement as well as plant construction and operation. The successful deployment of SMRs in developing countries in the future will require considerable reductions in requirements for these resources.

Underlying several of the challenges noted above is a lack of current agreement among decision-makers regarding the health effects of low level radiation. Research on radiation effects at the cellular level is providing new understandings, which may lead to more definitive criteria. Establishing general agreement on health effects and specified quantitative design guidance for limiting radiation exposure to plant personnel and the general public to a level consistent with other natural and man-made environmental risk factors is a necessary condition for meeting these challenges effectively.

3. DESIGN CONCEPTS

While many of the challenges are common across the full range of sizes, the small and medium size ranges tend to exhibit fundamental differences of approach to the designs.

Economic competitiveness in the small reactor range demands a more innovative approach to design, licensing and operation, and a higher reliance on standardization of design and mass production, forcing a more innovative departure from existing plants, use of waste heat for cogeneration, etc. Medium sized reactors tend to be more heavily based upon existing plants, focused primarily on electricity generation, and addressing challenges through effective use of experience and incorporation of evolutionary advances in technology and plant operations. However, the challenges that must be addressed, as well as elements of the required features, are similar for both size ranges.

Many SMR concepts have been developed in recent years. An IAEA TECDOC produced in 1995 identified a total of 44 concepts and provided information on the design and development status for 29 concepts. The names of the concepts identified in the TECDOC are shown in Figure 1. Work on some of these concepts has stopped, while others remain active and new concepts have emerged, as will be indicated in the concept description sessions of this SMR Seminar. In general, the concepts can be ordered by the following characteristics:

- **Fuel** – Fuel forms include oxides of uranium and plutonium as cylindrical pellets in thin walled tubing or as ceramic coated micro particles in a graphite matrix, fast reactor concepts also include nitride as well as metal fuels.
- **Moderator** – The moderator materials (used to thermalize the neutrons and thus not present for fast reactors), which can also serve as the coolant fluid, include light and heavy water as well as graphite.
- **Coolant** – Coolant fluids include light and heavy water, helium, and liquid metals (sodium, lead and lead/bismuth alloy)
- **Cycle** – The thermal energy is converted to mechanical (and electrical) energy through the use of a Rankine cycle with steam as the working fluid, or a Brayton cycle with helium as the working fluid. An indirect cycle incorporates a heat exchanger between the reactor core and the power turbine.
- **Configuration** – The system can be characterized as having an integral (reactor core and heat exchangers in a single vessel) or loop (multiple vessels connected by piping) configuration.

The range of combinations of these features, and their applicability to the small and medium size ranges, is shown in Table 1.

4. FEATURES

The future nuclear energy marketplace in most countries can be expected to resemble the situation of the evolving marketplace for fossil energy use. In this environment, multinational power plant design and construction companies operating in conjunction with large multinational generating companies will likely dominate new generation. Local transmission and distribution organizations may execute long-term power purchase contracts with the generation companies, subject to the regulatory requirements of the host country, or purchase energy on a competitive wholesale market. The host country will likely recover investments in infrastructure and ongoing costs required for regulation through fees levied on plant operators. Successful SMRs are likely to have features that support an integrated, internationally standardized design, construction, operation and regulation regime in order to successfully penetrate the large portion of the future generation market expected in developing countries, and to avoid the need for a substantial investment of time and resources in the development of local nuclear industry infrastructure.

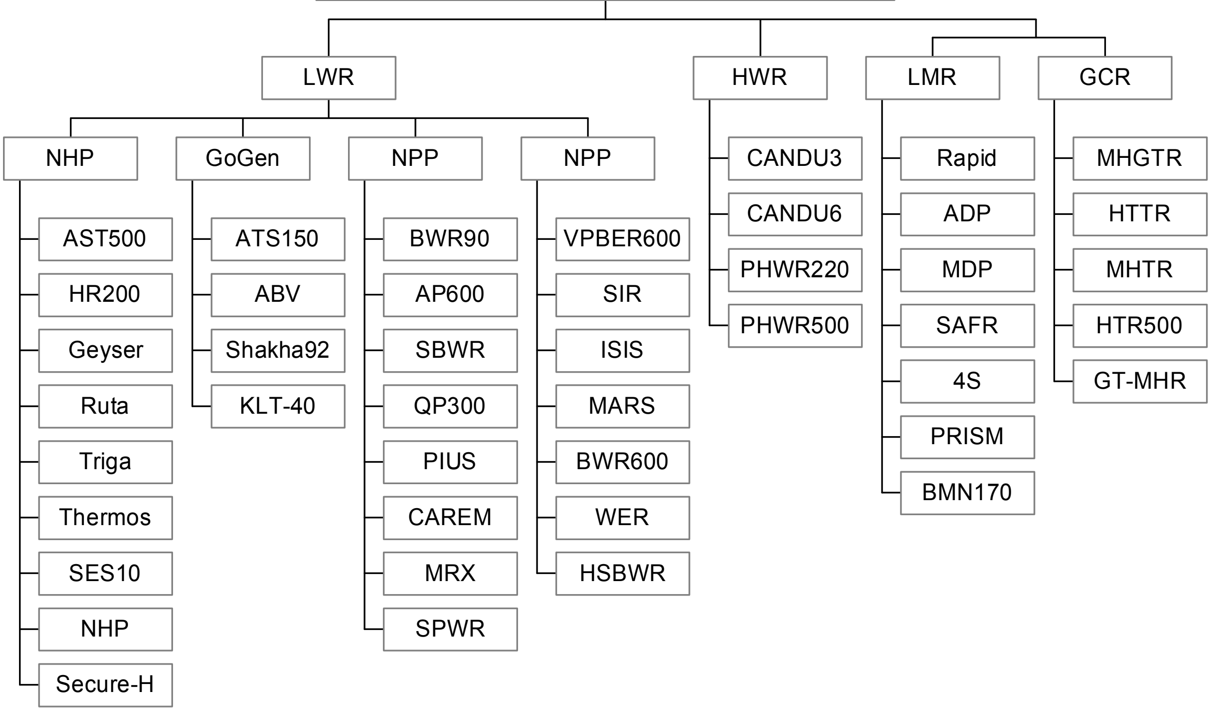
In light of the above considerations and the challenges identified earlier, the following desirable features for SMRs can be identified:

- **Proven, Standardized Design** – Deployment of a large number of units with essentially identical systems, structures and components is highly desirable. The performance characteristics should be known with high confidence based on testing and operation of an existing commercial or demonstration unit that does not differ from the standardized design in its significant features.
- **Simplified, Standardized Operation and Maintenance** – For smaller the unit sizes, smaller operation and maintenance staffs can help to maintain competitive cost. Normal operation, maintenance and surveillance activities can be simplified and standardized to minimize plant staff size as well as qualification and training requirements, to maximize use of indigenous craft and technician personnel and to support implementation of proven best practices and support software, with reliance on a central support organization for infrequent or unexpected activities.
- **Simplified, Standardized Regulatory Requirements** – Optimal use of inherent characteristics and passive design features can minimize and simplify the regulatory inspection and compliance requirements for initial unit acceptance and continued operation. An internationally accepted safety and licensing framework for unit design, construction and operation, including guidelines for inspection requirements and procedures, can minimize the necessary scope of host country regulatory infrastructure.
- **Short Construction Times** – Extensive use of factory fabricated modular systems and proven standardized construction practices to minimize the site labour force and construction time is highly desirable. This will provide benefits in generation planning as well as reduced cost and financial risk.
- **Stable, Flexible Normal Unit Operation** – It is desirable for the unit to be capable of operating through significant variations in grid frequency and voltage without tripping off the line. Load following capability over a substantial fraction of the unit power range is also desirable, recognizing that cost penalties associated with reduced unit capacity factor will likely limit its use to infrequent circumstances.
- **Stable, Slow Accident Response Characteristics** – Plant response to the full range of foreseeable accident conditions may be such as to minimize or eliminate need for rapid or complex manual or automatic mitigation actions to protect plant personnel or the public. Sufficient time (days) may then be available to engage offsite resources if necessary to respond to unforeseen conditions.
- **Flexible Siting** – In order to capture the value of smaller units with regard to reduced infrastructure and better matching with loads, it is desirable to locate the units within reasonable proximity of the loads.
- **Spent Fuel Disposition and Unit Decommissioning** – If provisions for disposition of spent fuel are well established, major liabilities or requirements for infrastructure development on the host country can be avoided. It is also beneficial if provisions and requirements for unit decommissioning are well defined and their costs factored into the economic competitiveness assessment.

TABLE I. VARIATIONS OF SMALL AND MEDIUM REACTOR DESIGNS

Fuel	Moderator	Coolant	Cycle	Configuration	Small	Medium
UO_2 Pellets in Pins	Light Water	Light Water	Heating Reactor	Integral		
			Direct Rankine (BWR)			
			Indirect Rankine (PWR)	Loop		
				Integral		
	Heavy Water		Direct Rankine (AHWR)			
		Heavy Water	Indirect Rankine PHWR)	Loop		
U-Pu Metal	none	Sodium	Indirect Rankine	Integral		
$(U\text{-}Pu)O_2$						
$(U\text{-}Pu)N_2$		Lead				
UO_2 Particles in spheres	Graphite	Helium	Indirect Rankine	Loop		
			Indirect Brayton			
			Direct Brayton			
UO_2 Particles in Blocks			Indirect Rankine			
			Direct Brayton			

Figure 1 SMR Development Lines in the World (1995)



- **Effective Safeguards Measures** – Simple and effective measures for safeguarding against diversion of special nuclear materials can minimize cost and promote confidence in the resolution of diversion and proliferation concerns.

Effective provision of these features in a given concept will require a well co-ordinated effort with regard to technology development, design specification, and design documentation addressing unit construction and regulatory interfaces, as well as necessary training materials and operating procedures. Identification of the design scope to be standardized, including allowed variations within that scope, if any, and interface requirements for the remainder of the unit will also be of considerable importance. And of course, all of this needs to be achieved while meeting the overall challenge of achieving competitive economic performance.

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INTERNATIONAL TRENDS AFFECTING THE DEVELOPMENT OF SMALL AND MEDIUM SIZED REACTORS

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Abstract

The beginning of the 21st century is characterised by significant changes in the energy policy-making framework of most countries. Globalisation of the world economy, deregulation of electricity markets, privatisation of the electricity sector, increasing concerns about the need to protect the environment and awareness of sustainable development goals are among the major trends affecting policy making and decisions in the energy sector. All those factors have impacts on nuclear energy programmes and may affect SMR development in particular. The paper investigates how those factors may change national energy policies and impact on nuclear energy programmes, with special emphasis on the potential role of SMRs in energy supply mixes. It elaborates on recent trends to increasing reliance on market mechanisms in the energy and electricity sectors and on the role of governments in implementing an integrated approach to policy making, within a sustainable development framework incorporating economic, social and environmental dimensions. Against this backdrop, the paper examines the potential markets for SMRs, taking into account their possible uses beyond electricity generation, such as potable water production, cogeneration, process or domestic heat supply and eventually hydrogen production. It reviews key issues to be addressed in order to facilitate the deployment of SMRs in different countries. Those include technology progress and transfer, capacity building in developing countries, adaptation to market requirements, economic competitiveness and social acceptance. The paper concludes with remarks regarding the importance of international co-operation, especially in the field of research and development on advanced reactor and fuel cycle concepts. In this connection, the role of intergovernmental organisations as facilitators and catalysts of national efforts is highlighted.

1. INTRODUCTION

It is a pleasure for me, at the outset of the Seminar, to share with you some thoughts on the opportunities and challenges for the development of small and medium reactors (SMRs). As I said during the Opening Ceremony, I think that this Seminar is very timely and my Agency is very pleased and honoured to co-sponsor this event. Changes in the decision-making landscape of the energy sector and renewed interest in research and development of innovative reactor concepts are key factors that make it relevant to re-assess the future role of SMRs in the energy supply mixes for the 21st century.

Although a large majority of the nuclear power plants in operation in the world are large size reactors, there has been over time a continued interest for SMRs in many countries. As a result of research and development programmes undertaken in those countries, a significant number of reactor designs within the small and medium size range, i.e., below 700 MWe, have reached different degrees of achievement and could be commercially deployed provided there would be a demand from the market. Indeed, the number of research and design teams represented in the Seminar demonstrate the variety and liveliness of development activities in the field.

International trends to electricity market deregulation, leading to a growing number of independent producers and the development of distributed generation, are likely to strengthen the attractiveness of small and medium size power plants. Modularity and flexibility will become major advantages in liberalised markets where customers are not captive and, therefore, future demand is uncertain. SMRs having lower total investments than large size reactors should be less affected by the reluctance of private investors to accept the risks of capital intensive projects with very long pay back periods. SMRs, however, will have to demonstrate their industrial feasibility and economic competitiveness before their entry in the market could be considered.

The presentations that we will hear later in the different sessions of the Seminar will review and analyse in detail various technical, economic and safety aspects of SMRs. As a backdrop to further discussions, I would like to review briefly the evolution of policy-making factors in the energy field, investigating how this evolution may affect the development of SMRs, and to elaborate on the potential markets for SMRs, taking into account their specific characteristics and the challenges for their deployment. In my concluding remarks, I will highlight the importance of international co-operation for strengthening the efficiency of national efforts in the area, with emphasis on the role of intergovernmental organisations.

2. FACTORS AFFECTING ENERGY POLICY MAKING

The beginning of the 21st century is characterised by significant changes in the energy policy-making framework of most countries. Globalisation of the world economy, deregulation of electricity markets, privatisation of the electricity sector, increasing concerns about the need to protect the environment and awareness of sustainable development goals are among the major trends affecting policy making and decisions in the energy sector. All those factors have impacts on nuclear energy programmes and may affect SMR development in particular.

National energy policies are based upon country specific contexts and priorities but the main driving factors in energy policy making are similar world-wide. The evolution of energy supply mixes and the rate of change between alternative sources or technologies have been driven by a limited number of factors relating to economic development and competitiveness, as well as social and environmental protection issues. Although recent trends place emphasis on market mechanisms to ensure competitiveness, governments, especially in OECD countries, are increasingly considering an integrated approach to policy making, within a sustainable development framework, incorporating economic, social and environmental dimensions.

Energy demand, obviously, is a driving factor in supply policies. The analysis of future energy demand characteristics indicates that the types of energy services that society will look for in the 21st century will differ from present uses and needs both in quantity and in quality. Population growth and trends to urbanisation will increase total energy demand and the need for large size supply sources. On the other hand, the deployment of computers and of telecommunication technologies, in particular based on the internet, is leading to greater decentralisation of the workplace and calling for distributed supply sources rather than for centralised electricity generation. The evolution towards service supply is a major trend in the energy sector, calling for flexibility and adaptability of generation sources, that will affect technology choices. In this regard, power plants of moderate size, such as SMRs, offer a number of advantages.

Competitiveness remains a cornerstone in energy policy making but the framework within which comparative economic assessments are conducted is evolving. Increasing emphasis is placed on market mechanisms for promoting optimised energy supply mixes, in particular for electricity generation. Economic deregulation introduces competition throughout the electricity sector, and privatisation is changing the criteria used by generators in their technology choices.

The key factors affecting the economic competitiveness of alternative electricity generation sources and technologies are essentially fossil fuel prices, capital costs, expected rates of return on investments, and technological performance, e.g., thermal efficiency, availability factor and technical lifetime of power plants. The volatility of hydrocarbon prices, that has been demonstrated again recently, is challenging the economic advantages of fossil-fuelled power plants. In particular, combined cycle gas turbines, which are the main competitor for SMRs at present, may lose most of their competitive margin if gas prices continue to rise.

Open competition in the electricity sector should eventually eliminate captive markets, upon which utilities that enjoyed a monopoly used to rely on for guaranteed future demand. This increases the financial risk associated with capital intensive technologies. In this context, SMRs may become more attractive because of their short construction time and relatively low investment costs, as compared with large coal-fired or nuclear power units. Their investment costs can be amortised rather quickly and, therefore, the financial risk imposed on the generating company shareholders is reduced significantly.

The cost of capital depends largely on the rate of return expected by investors. Private investors that are becoming key actors in the electricity sector generally require high financial returns and this may modify the ranking of electricity generation options. For capital intensive technologies such as nuclear power, the impact of higher rates of return on total generation costs is adverse and very significant. Within alternative nuclear power plant options, however, high discount rates affect less SMRs than large size units because the former have shorter construction periods.

The increasing awareness of environmental issues and more broadly the sustainability goals, including long-term security of supply and protection of people and eco-systems, are giving a stronger weight to non-economic criteria in energy policy making. Explicitly integrating the concept of sustainable development in energy policies is calling for strategies that preserve natural resources and the environment, reduce regional disparities and give equal opportunities to present and future generations world-wide.

The extent to which higher priority on environmental protection and long-term sustainability will affect energy and electricity policies is difficult to predict and the outcomes are likely to differ from country to country and over time. Uncertainties over the impacts of energy-related environmental burdens remain quite large, in particular with regard to greenhouse gas emissions and the threat of global climate change. However, a clear commitment emerges at the international level to tackle ways to control CO_2 emissions from industrial activities, even if views on the most appropriate policies and measures in this regard may differ from country to country. Therefore, energy sources having lower emissions are likely to be chosen whenever technically feasible and economically competitive.

From this point of view, nuclear energy, including SMRs, may be viewed as a key option to implement in sustainable energy supply mixes because it is a carbon-free energy source that relies on plentiful natural resources, uranium and thorium, that have no other significant commercial use. On the other hand, concerns raised by long-lived radioactive waste disposal and low probability/high consequence nuclear accidents are social and political hurdles that need to be overcome in order to secure a future role for nuclear energy and possibilities for the deployment of SMRs.

3. POTENTIAL MARKETS FOR SMALL AND MEDIUM REACTORS

Small and medium size reactors are adapted to decentralised energy demand and their deployment may be feasible in various market conditions where large nuclear units would have difficulties to compete. Generally, in order to ensure grid stability the size of the largest unit on a network should not exceed 10% of the total interconnected capacity. The trend to industrialisation and urbanisation in developing countries increases the demand for electricity in regions where grids are rather small. SMRs are well adapted to those circumstances where the introduction of large nuclear units would not be possible.

In OECD countries, the option of decentralised electricity generation is considered with increasing interest in the context of market deregulation and privatisation. The growing importance of independent power producers (IPPs) favours the implementation of medium size power plants that are easier to finance and manage and represent a lower risk in terms of securing a market for the electricity generated.

Nuclear fission reactors have demonstrated already their ability to supply electricity reliably and economically, but the future of nuclear energy will depend not only on electrical applications. Worldwide, the share of electricity in final energy consumption is around 20% while over half of the demand for energy is related to heat supply, including steam, process heat and hot water for domestic and industrial applications. Therefore, the design and development of nuclear reactor concepts adapted to non-electrical uses is essential for future deployment of nuclear energy.

Although only a few nuclear units in operation in the world are used for district and/or process heat generation, they demonstrate that other applications of nuclear energy are already possible. In the future, the energy produced by nuclear reactors may be used also for potable water production, a service for which demand is expected to increase rapidly in many regions. Finally, nuclear energy could play a significant role for large-scale production of hydrogen, if and when hydrogen would become a major secondary energy carrier.

Small reactors can be used to supply heat and electricity, or heat only, for industrial applications and district heating. Their advantages over large nuclear units for this type of application, besides the better adaptation of their size to the demand, are linked with safety characteristics allowing their construction in densely populated areas. Nevertheless, finding adequate sites and gaining public acceptance may remain difficult as well as reaching competitiveness with alternative options.

For a number of industrial heat supply applications, e.g., for coal, oil and tar sand processing, and for hydrogen production, the required temperatures range from 300 to 1 000 ºC and the technology of choice seems to be High Temperature Gas Cooled Reactors (HTGRs). The modular approach adopted in most advanced HTGR designs provides opportunities for reducing costs through series effect compensating for the lack of economy of scale resulting from small sizes.

The use of SMRs for sea water desalination deserves specific attention in the light of its importance in a number of countries where potable water shortages are already experienced or expected to occur in the near future. An entire session of this Seminar is devoted to nuclear desalination, therefore I will not elaborate in detail on the potential market in this area. I would like to stress, however, that competitiveness with alternatives, safety and reactor design aiming at user friendly operation and maintenance are very important characteristics for this application.

4. CHALLENGES FOR SMR DEPLOYMENT

While the prospects for the development of SMRs are promising, they face a number of challenges that are either generic to nuclear energy systems or specific to small size units. Most of the key barriers to a larger deployment of nuclear energy in general, including infrastructure requirements, economic and financial risks, and lack of public acceptance, apply to SMRs. Since most SMR concepts under development have adopted innovative approaches to design and safety features, it may be expected that designers will endeavour to address those issues in a satisfactory manner.

Establishing adequate infrastructures to support the construction and operation of SMRs may be difficult in some countries where this type of reactor would be otherwise an attractive option. However, in most cases, the designs and characteristics of SMRs should facilitate their implementation in countries with limited experience in the field of nuclear power deployment. Adaptation and transfer of technology and know-how will be essential in this regard but localisation should be facilitated by the relative simplicity of designs and the potential for using factory production.

Regarding economic and financial aspects, total capital cost is lower for SMRs than for large size units, even if their specific cost per MWe installed is generally higher. Therefore, SMRs are likely to be easier to finance by private investors and/or in countries with limited capital availability. It remains that competitiveness with alternatives, including fossil-fuelled power plants, renewable energy sources and large nuclear units, will be a prerequisite for the deployment of SMRs. In order to compensate for the lack of economy of scale, designs should place emphasis on simplification and modularity allowing for fabrication in series of most elements of the plants. Shortening of construction time could be a key factor in reducing the total capital cost and thereby enhancing competitiveness.

The evolution of decision-making processes for all industrial projects is characterised by an increasing involvement of various stakeholders. Key features of SMRs include simplification and streamlining of designs as well as emphasis placed on safety features avoiding off-site impacts in case of accident. Such characteristics should facilitate their acceptability by local communities. However, a large development of SMRs would require their construction on many new sites and may raise issues of public acceptance in particular in small rural communities or protected areas.

In spite of the demonstrated effectiveness of the international safeguards regime, the risk of proliferation of nuclear weapons remains a social concern deserving to be addressed by governments and the nuclear industry. A significant deployment of SMRs would lead to building a large number of reactors in many different countries and sites. Therefore, gaining social acceptance will require specific efforts of designers to enhance the proliferation resistance characteristics of SMRs.

CONCLUDING REMARKS

Energy markets of the 21st century will be challenging for all technologies and emerging options such as SMRs may have difficulties to enter in a commercial phase in the light of the emphasis placed by investors on short-term benefits. However, governments of countries wishing to keep the nuclear option in the framework of sustainable energy mixes for the future may consider supporting further research and development on small and medium size reactor concepts.

In order to reach the stage of potential commercial deployment, most SMR concepts need additional efforts aiming at demonstrating fully their technical feasibility and economic competitiveness. Strengthening international co-operation in this field could enhance the efficiency of national programmes and reduce the costs to be borne by each country. International initiatives such as “Generation IV” and “INPRO” are to be commended in this regard.

The role of intergovernmental organisations, such as the IAEA and the NEA, is essential in providing countries with a framework to undertake jointly projects that benefit from the synergy of multi-national teams and expertise. Intergovernmental organisations also play a key role in offering a forum for exchange of information and sharing of experience between countries. This Seminar is an example of such a forum and I am sure that the presentations and discussions of the coming days will be fruitful and lead to valuable findings and conclusions.

NEXT STEPS IN THE EVOLUTION OF NUCLEAR POWER TECHNOLOGY: THE POTENTIAL OF SMALL REACTORS

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Abstract

This paper addresses current and planned DOE technology development programs and some of the candidate technologies. Small and medium reactor concepts being explored under the Nuclear Energy Research Initiative (NERI) are identified, and a recent study of very small reactor concepts is summarized. SMR concepts are discussed in terms of the goals of improved economics, proliferation resistance, safety and waste minimization.

1. INTRODUCTION

The United States, with its large population and well-developed electric power grid, has historically opted to build ever larger power plants, whether coal or nuclear. Recently, some factors have caused a reexamination of that trend, but there is presently no consensus as far as whether future power plants will continue to be large, or whether smaller plants will become attractive.

The current program for developing the next generation of nuclear power plants is in its early stages, and has as yet placed no constraints on the types of technologies or sizes of units that should be developed. This paper will, therefore, outline the overall US program, and highlight some of the issues regarding small reactors that are likely to be considered in the course of the reactor development program.

2. THE NEED FOR NUCLEAR POWER

The demand for electricity continues to grow, both in the United States and worldwide. The demand is greatest in the developing world, which still has large populations with little or no access to electricity. The recognition that electrification fosters economic development and brings a host of benefits to the population has increased the emphasis developing countries are placing on building power plants and electric grids. The United States, of course, is already a heavy user of electric power, but growing use of electronic media, continuing population growth, and other factors have caused a continuing modest increase in demand.

At the same time, there is a growing concern, worldwide, about the impacts of industrialization on the environment. Global warming, with its possible significant long-term impacts on the world, is of particular concern. This concern has raised warning flags about the impacts of fueling further industrialization with carbon-based fuels, and is creating pressure to look more seriously at all alternatives. There is also widespread recognition that renewable energy sources, while they are attractive alternatives in some cases, will not be able to meet needs of the magnitude expected in the coming years.

In the United States, several additional factors are also resulting in renewed interest in nuclear power. The deregulation and restructuring of the electric utility industry has created a number of economic factors that favor reconsideration of nuclear power. First, the consolidation of utilities has resulted in a few utilities with large numbers of nuclear units. The size of the utilities and the number of plants each owns result in economies of scale in operations, and in the ability of management to devote adequate resources and to develop appropriate expertise to manage a fleet of nuclear power plants. Secondly, the provisions of deregulation have created a market for the purchase and sale of single unit plants.

During this same period, the US Nuclear Regulatory Commission (USNRC), which is responsible for oversight of the US commercial reactors, has adopted a more risk-focused approach to regulation. This has provided greater regulatory stability for the nuclear operators, which has further encouraged utilities to continue operating plants. At the same time, initiatives within the industry have resulted in improved plant operations, resulting in greater power production—the equivalent of about 25 new plants--and greater cost-effectiveness. Among its recent accomplishments, the USNRC recently relicensed several plants for 20 years of operation beyond their initial 40 year licenses. This environment permits utilities to plan with greater certainty to make significant capital investments that will allow the plants to perform at higher efficiency.

More recently, dramatic price increases in natural gas and electricity shortages in the State of California, have also contributed to a growing recognition of a need to build new power plants, and to broaden our options beyond the combined-cycle gas-turbine plants that have, in recent years, been the option of choice.

On May 17, 2001, the new recognition of the need for nuclear power was enunciated by the Office of the President in the Vice President's National Energy Policy Report. That report, of course, addressed all areas of energy supply and demand, including fossil fuels, renewable energy, conservation, and nuclear power. In the latter area, it supported the expansion of nuclear energy in the United States as a major component of our national energy plan. It endorsed a wide range of options designed to make fuller use of our current inventory of nuclear power plants and to develop the next generation technologies and fuel cycles.

3. CURRENT AND FUTURE REACTOR DEVELOPMENT PROGRAMS

The US Department of Energy (DOE) already has in place a number of programs that are consistent with the goals outlined in the Vice President's report. These may form the core of any future programs that result from the implementation of the National Energy Policy Report. They include programs aimed at increasing the production from existing reactors, relicensing existing reactors, demonstrating the licensing process for existing advanced reactor designs, and developing new reactors and fuel cycles. The remainder of this paper will focus on existing and potential activities in the latter area.

We have come to call the future nuclear power plant technology "Generation IV" (see figure 1). This terminology arises in the following way: We consider Generation I to be the first commercial power reactors. These were prototypes and demonstration plants that have now been shut down. The next generation, Generation IV, followed. These are basically the plants that are currently in operation in the United States and worldwide. The next designs after that were the advanced light water designs that are now beginning to be built and operated in some countries. These, we call Generation III reactors.

After that, we must speculate about what comes next. Most people think there will be some evolutionary designs--improvements on existing reactors--that can be deployed in the next decade or so. These might be called Generation III+. However, the next true generation of reactors, Generation IV, is anticipated to embody significant improvements over existing designs. We do not yet know what technology such reactors will use, whether they will be large or small, or what fuel cycle they will use, but we expect that they will offer ways to overcome current commercial and public acceptance impediments to further nuclear deployment. In particular, we anticipate that they will be highly economical, have enhanced safety characteristics, minimize waste production, and be highly proliferation resistant. We project that significant research will be needed to develop such technologies, and that they would be deployable by about the year 2030.

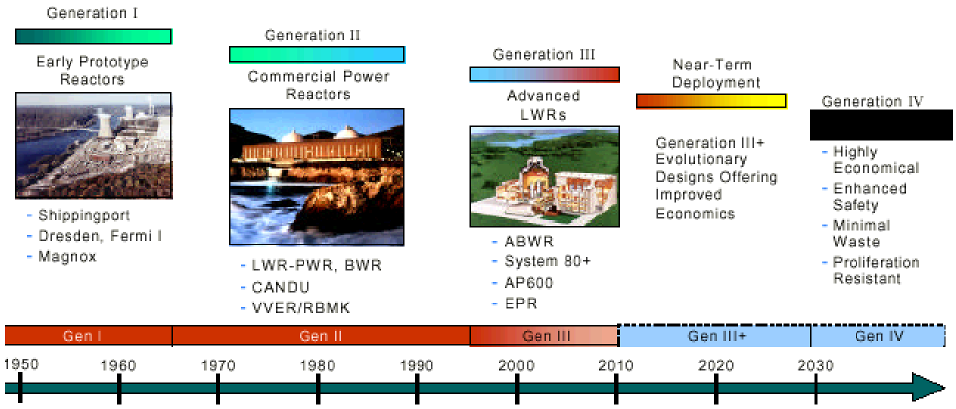

FIG. 1. The Evolution of Nuclear Power.

The US Department of Energy has established a three-pronged program aimed at the development of these Generation IV reactors in partnership with other countries. The three parts of the program are: a roadmap development effort, a research program, and an international forum for cooperation. Other papers will describe each of these elements in greater detail, but briefly, these elements encompass the following activities:

Roadmap: One major effort is the development of a roadmap for nuclear technology development. This is a two-year effort designed to identify the most promising technologies and to develop a plan to identify the research and development needed to bring the most promising concepts to a point where they can be deployed. Concepts from around the world will be considered, and the teams that are conducting the evaluations are drawn from an international community of experts in all technology areas. Research: The United States currently has a small research effort, called the Nuclear Energy Research Initiative (NERI), to explore a wide range of innovative nuclear technology concepts. We are also partnering with other countries in an international version of this program called International NERI (I-NERI), which features bilateral arrangements with other countries. Both of these programs support small one to three year, investigator-initiated, peer reviewed projects in reactor design concepts, fuel cycle, and basic nuclear research. Most of these projects are highly collaborative, with US national laboratories, universities, and industrial organizations participating. There are over 50 projects now, and another dozen or so will be funded in 2001. Organizations from other countries also participate in some NERI projects, and international participation will be a requirement for I-NERI projects. Early results from these projects will feed into the roadmap development. After the roadmap has been completed, larger research programs may be initiated.

International Cooperation: The third element of our effort is the establishment of a Generation IV International Forum (GIF), currently involving nine countries, that exists to collaborate on the development of Generation IV technologies. Members of the GIF are participating in the roadmap development, and plans are being developed for international research and development collaborations. Initial efforts are largely on I-NERI programs, but plans are being developed for larger collaborations, which may be focused on specific technologies and may be multi-lateral and long term in nature. The GIF will continue to exist during the research phase in order to monitor and provide guidance to the Generation IV development.

There are, at present, no constraints on the range of technologies under consideration. Concepts being collected for evaluation, and concepts being explored in NERI projects, include the full range of reactor technologies (water, gas, liquid metal, and non-classical), alternative fuel cycles (including uranium and thorium), different applications (electricity production, process heat, desalinization, and hydrogen production), and different sizes (small, medium, and large).

4. THE QUESTION OF SIZE

Traditionally, the trend in the United States and most other countries has been to larger nuclear power plants. Larger plants appeared to offer the best economies of scale. Recently, however, new attention is being given to the potential benefits of smaller sized nuclear power units. Some of the advantages are economic benefits that may offset the economies of scale of larger plants. For example, smaller plants can be factory-fabricated, and the production of multiples of a single design at a factory site offers some cost advantages. It may also be possible to better match the growth of supply to the growth of demand, and avoid the necessity of building large amounts of excess capacity in expectation of future demand growth.

In addition, smaller plants may have other benefits. For remote applications and harsh climates, the economics are much different, and a smaller nuclear unit with a long refueling interval may offer advantages over alternative technologies that require frequent shipments of fuel. Smaller units may also be suitable for applications other than electricity production (district heating, process heat, hydrogen, desalinated water), and the potential for co-production of electricity and other products may make such units particularly attractive in areas where there are demands for the other products. In general, in the United States, very small reactors appear to be attractive for special needs. However, outside the United States, in areas of the world where populations are sparser or electric power grids are not well developed, there may be considerable demand for small nuclear power plants.

Clearly, the attractiveness of smaller nuclear plants is based on a number of premises which remain to be demonstrated. For the continental United States, where large plants have been the norm, it remains to be seen whether the economies of factory fabrication will outweigh the traditional economies of scale. In areas of the United States where it is presumed that very small plants may be attractive, such as remote parts of Alaska or Hawaii, the characteristics likely to be needed for remote application, such as long refueling intervals, will have to be developed and demonstrated, and the markets in those areas (including public acceptance considerations) will have to be confirmed.

A number of technical and regulatory challenges will also need to be addressed. Many of the concepts for smaller reactors are based on technologies with more inherent safety features, and some of the economies are predicated on the fact that fewer emergency safety systems may be needed, and that full containments may not be needed. These characteristics obviously need to be demonstrated and need to satisfy regulatory requirements. Other features, such as long refueling cycles, also must be demonstrated. In remote areas, the plants must be able to operate with limited technical staff, and the physical security of the plants must be assured.

It is also clear that there are several potential concepts for small reactors, each of which may require reactors of different sizes. Remote applications and some developing country needs may require very small plants, other developing country needs may require mid-sized plants, and nuclear parks in industrialized countries may require mid-sized or larger plants.

5. CURRENT US ACTIVITIES

As previously noted, there are no US government programs that are aimed exclusively at small and medium reactor concepts. However, all the technology development programs include all size ranges in the concepts being considered. Thus, the roadmap will review all technologies, and design concepts in all size ranges. Also, the current NERI program includes projects for both large and small reactor concepts. Another paper will describe NERI projects and other small reactors concepts being developed or studied under DOE programs at present.

In addition, private industry has several initiatives underway related to SMRs. In particular, one US utility (Exelon Corporation), has an equity share in a South African 114 MWe pebble bed modular reactor (PBMR). Both General Atomics and Westinghouse have design concepts for small and medium-sized reactors under development. (In addition to corporate funding, these efforts have some NERI or other government support, as well as involvement by other organizations.)

Finally, DOE is currently completely a report to Congress on small reactor technologies and their feasibility for deployment in the United States. Drawing on technologies under development worldwide, this study has identified a number of very small reactor concepts that may be potentially deployable in the relatively near term, has done very preliminary studies of the status of technical development and the potential regulatory issues, and has identified some of the economic considerations. Preliminary results suggest that some technologies may be competitive candidates for application in remote areas of the United States, such as Alaska and Hawaii.

6. A VISION OF THE FUTURE

It is clear that in the United States and other industrialized countries, the demand for electricity will support the continued deployment of large reactors in many cases. However, it appears that applications for smaller reactors likely exist in the United States, and certainly exist worldwide. One might envision a future where a range of reactor sizes will be available for deployment for different locations and different applications. For example, large nuclear power plants may continue to be built to serve large populations and areas with well-developed electric power grids. At the same time, nuclear parks may be developed to house ten or twenty small or medium reactors (perhaps of the order of about 100 MWe), which could be constructed over a period of time as demand in the area grew. World-wide, single reactors of the same size range might serve smaller populations or sparse populations with well-developed grids to deliver the power throughout a region. And in very remote areas, very small reactors (perhaps about 10 to 50 MWe) might be deployed to small, isolated communities. Depending on the need, reactors in any of the sizes may be configured provide more than one output, such as desalinization, process heat, district heating, or hydrogen production. Each size range and each application has somewhat different requirements and therefore may employ different nuclear technologies. As a result, the future fleet of reactors worldwide could include both water-cooled designs and other design concepts. Thus, the research program we have started in the United States, which encompasses different sizes and different technologies, is well designed to address multiple future needs. These needs will be kept in mind as the roadmap effort narrows the consideration to a smaller number of designs.

SMR FOR DESALINATION OF SEAWATER: POTENTIAL AND CHALLENGES

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Abstract

The demand for fresh water and electricity will continue to rise over the next decades. These are driven by four main factors: population growth, economic development, technological advancement, and environmental related aspects. One of the most promising approaches for securing abundant fresh water is seawater desalination and the best energy source (for electricity and/or heat) is a small or medium sized nuclear reactor if we are also looking for reduction in the level of GHG and CO_2. Seawater is inexhaustibly available, and desalination technologies are well developed. About three decades of experience have been gained in the use of nuclear desalination, and several countries are actively involved in developing this technology and are moving towards implementing a nuclear desalination program. International co-operation in nuclear desalination is not only following the new trends all over the world, but it also provides the best means to ensure safe and reliable application of nuclear energy.

1. INTRODUCTION

Adequate supply of electric energy and potable water for domestic as well as industrial and public uses is one of the fundamental conditions for development, and indeed one of the major challenges. However, while energy is the engine to improve the quality of life and increase living standard, fresh water is an essential element for human life.

Nature has bestowed on mankind primary energy in various forms of which nuclear energy is the friendliest. Until recently, about 90% of the world energy consumption consisted of fossil fuels, which are the main source of pollution as well as emissions of CO_2 and Green House Gases (GHG). Nuclear energy does not generate any of these gases. The main sources of fresh water in the world are rivers and underground waters. The share of desalted seawater is still very limited, but it is increasing steadily to compensate for the increasing deficit in fresh water.

There is no doubt that the demand for fresh water and energy, particularly in the form of electricity, will continue to rise over the next decades. As a minimum, the supply of energy will need to double over the next 40 to 50 years, and the demand for electricity and fresh water is likely to increase even more, and some projections indicate that it will double as early as 2030. Clearly, there is a need to identify the potential and challenges to supply electrical energy and fresh water to meet the future needs in a sustainable manner and to achieve “Minimum Regrets” options, following the universal environmental requirements. The United Nations designates the 21st Century as the Century for Water. The big questions are: what is the role of NPP, in general, and SMR in particular, in producing potable water? And what is potential for SMR and what are the challenges? In reality the problem has three dimensions, namely:

- Water shortage,
- Energy source, and
- Clean environment.

FACTORS AFFECTING GLOBAL WATER AND ENERGY DEMAND

Water and energy demands are driven by four main factors:

1. Population growth,
2. Economic development,
3. Technological advancement, and
4. Environmental related aspects

1.1. Population growth

Water and energy consumption will depend to a great extent on population growth and urbanization. Forecasts show that the world's population will increase from about 5.5 billion by the end of the 20^{th} century to about 8.5 billion in the year 2030. Moreover, the population movement from rural to urban areas and changing life style will have an impact on demand. It is projected that urban population will constitute about 65% of the total population in the year 2030 compared to 46% in the year 2000. The average world population growth rate is about 1.4%, and in the developing countries, where the need for fresh water and energy are much more, the population growth rate is even higher, e.g., in South Asia it is 1.7%.

1.2. Economic development

Both energy and water are essential for the development of societies and nations. The expected global average GDP growth per year is about 3.2%, with 2.5% for OECD and up to 7.0% for certain countries, and this simply means more energy and fresh water will be needed.

1.3. Technological advancement

Since 1973, the world has saved more energy through improved efficiency than the contributions it has gained from all new energy sources combined. Several technological achievements have been realized, or are under development for electricity generation and seawater desalination. Current studies indicate that advancement in renewable energy could be realized soon. However, the only available source to generate electricity continuously (as base load) and reliably is nuclear energy.

1.4. Environmental related aspects

In recent years the world's concerns regarding environmental impacts have increased. The UNFCCC (United Nations Framework Convention on Climate Change) set a target in 1992 of reducing annual GHG emissions to the 1990 level by the year 2000. This was followed in December 1997 by the Kyoto protocol which, proposes a reduction in emission levels in developed countries to an average 5.2% below the 1990 level by the period 2008-12.

The growing and even more challenging concern for the world today is mitigating the effect of GHG and CO_2. The threat of climate change has compelled the international community to re-investigate the role of non-fossil energy sources such as nuclear energy. In this regard, nuclear energy is a prime candidate as the most reliable base load source, where the reliability of renewable sources such as solar and wind is much lower.

2. WORLD RESOURCES

In December 2000 the total estimated proven fossil fuel reserves were 140.4, 130 and 691 Billion TOE for oil, natural gas and coal, respectively, and in percentage they were 15, 14 and 71%. The consumption of oil, gas and coal in the year 2000 was about 3.50, 2.10 and 2.20 Billion TOE, respectively. Accordingly, the estimated proven reserves to production ratios were 40, 60 and 230 years. The studies show that during the last years no considerable proven reserves had been added, on the contrary the ratio of estimated proven reserves to production ratio was constant or even decreased. The worldwide consumption from these three fossil fuels represents about 89% of total global consumption. The nuclear share is about 8% and hydro about 3%.

The total water on the earth is about 1.4×10^{18} ton, of which 98.3% is seawater. Sweet water amounts to about 24×10^{15} ton, of which only 3% is fresh water, i.e., about 7×10^{14} tons. Now, the average fresh water per person is about 120×10^{3} ton. It will be about 60×10^{3} ton in the year 2030. However, both water and energy resources are unevenly distributed all over the world. Many places are limited or poor with these resources.

3. WATER SHORTAGE AND POSSIBLE SOLUTION

Mankind is expected to be faced with a shortage of this important resource, and some regions are already confronted with severe problems of fresh water shortages. Many countries have been making every possible effort to solve the fresh water shortage problem. Some of the practical approaches for solving the water shortage problem are: better use of natural resources (construction of multi purpose dams, further exploitation of underground water), better water management (recycling, improvement of water resources quality control), and water production (artificial rains, seawater desalination). Among those various approaches, one of the most promising approaches for securing abundant fresh water is seawater desalination. Seawater is inexhaustibly available, and desalination technologies are well developed.

4. DESALINATION OF SEAWATER

The basic principle of seawater desalination is to remove salt from seawater and bring down the dissolved solids to an acceptable level. Among the various commercially available large-scale seawater desalination processes are the distillation technologies and the RO membrane process. The basic mechanism for distillation methods such as Multi-Stage Flash (MSF) and Multi-Effect Distillation (MED) is to heat up the feed water, produce steam through boiling or flashing and then condense the steam to produce fresh water. The Reverse Osmosis (RO) membrane process applies mechanical pressure to separate fresh water from seawater. These technologies were developed in the 1960s and experience has been gained, worldwide. The total installed capacity of desalination plants has steadily increased to more than twenty million tons per day over the past four decades.

In desalting seawater, energy is generally supplied for the desalination process in the form of either steam and/or electricity. Conventional fossil fuels have been mostly utilized as energy sources to date for the existing desalination plants. These energy sources are expected to remain as major sources for the time being. On the other hand, the negative aspects in using these conventional fossil fuels have been also widely recognized. Typical negative aspects include air pollution, global warming due to the greenhouse effect, and depletion of useful natural resources. To cope with these negative aspects, alternative energy sources have been investigated with respect to availability, economical competitiveness, and scale. Among several possible alternatives, nuclear energy is favored because it is practical, available on a large-scale, and resource saving. Nuclear power plants remain the primary large scale, base load energy sources available to meet the world's ever increasing electricity and/or heat needs, while reducing or stabilizing GHG emissions.

5. EXPERIENCE WITH NUCLEAR DESALINATION

About three decades of experience have been gained in the use of nuclear energy for desalination since the Aktau plant in Kazakhstan first came into operation in 1973. The plant produces about 80,000 tons of fresh water per day for municipal and industrial uses. Also in Japan, about 100 reactor years of experience in nuclear seawater desalination have been accumulated since the 1970s at several nuclear power plants, although production is in small amounts in the range of 1,000 - 3,000 tons per day for in-plant use.

These examples have clearly shown that the application of nuclear energy to desalination does not raise technical impediments or cause safety-related incidents, and have demonstrated the practical utilization of nuclear energy for seawater desalination.

6. PROSPECTS FOR NUCLEAR DESALINATION

In response to the increasing interest in nuclear desalination and to its great potential, several countries have been actively involved in associated technology development and are moving towards implementing a nuclear desalination program. Morocco and China studied a nuclear desalination plant capable of producing 8,000 tons of fresh water per day from seawater by utilizing a nuclear heating reactor at a Moroccan site on the Atlantic Ocean. India launched a plan to produce 6,300 tons of fresh water per day by combining a hybrid desalination process to an existing nuclear power plant, and the civil work for the combination is currently in progress. The Republic of Korea has also been actively running a nuclear desalination program since 1997. The Korean program is currently focused on the development of a small nuclear reactor (SMART) and an integrated nuclear desalination system for producing about 40,000 tons of fresh water per day as well as electricity. The Russian Federation and Canada are in co-operation to develop a barge-mounted nuclear desalination system using technology of the KLT-40 reactor. Many other countries are also continuously showing their strong interest in proceeding with nuclear desalination, e.g.: Egypt, China, and Argentina. These countries are either studying nuclear desalination for national water supply or are developing appropriated reactors in the small and medium category for energy supply to desalination processes. These activities and efforts are based on the prospects of the use of nuclear energy for the seawater desalination. At the Agency's Symposium held in Taejon, Republic of Korea, in May 1997, the majority of participants shared the view that nuclear desalination is technically and economically feasible. Public acceptance and confirmation of cost viability were recognized as the remaining major issues.

The next challenging step would be to demonstrate nuclear seawater desalination for practical applications. The implementation of a nuclear desalination demonstration project would provide very useful support to the promotion of the commercial deployment of nuclear desalination plants. Since nuclear reactor technologies and desalination technologies have been well established, relevant issues for commercial application such as technical, economical, safety, infrastructure, and institutional aspects should be resolved. In addition, in-depth economic assessments of nuclear desalination plants are also needed in comparison with fossil-fueled desalination options. The demonstration project should also confirm that the nuclear desalination plant could produce fresh water economically.

Some technical issues, in particular those design features, which have a major impact on the overall economics of nuclear desalination, need to be demonstrated in order to confirm assumptions and estimates used in evaluations. These issues include the optimal coupling of the nuclear energy system and the desalination facility, an advanced reactor design if selected for coupling with desalination systems, performance improvement of the desalination systems, advanced desalination technologies, and the reliability of the integrated nuclear desalination system.

The radioactivity carry-over to the product water could be a critical issue for the practical implementation of nuclear seawater desalination. However, this issue has not been either reported from experience nor does it appear to be significant. To avoid any possible contamination of the product water, safety implications of coupling nuclear power plants with desalination units should be examined. Technical issues related to the radiological contamination include analysis techniques for the assessment of radiological impacts on the public, and evaluation of emergency preparedness measures, etc. Securing a higher performance of the desalination system will also contribute to ease safety concerns. These technical subjects of common interest are currently being tackled in the IAEA Coordinated Research Program with about ten participating institutes from the Member States.

7. SMR FOR NUCLEAR DESALINATION

It is clear that in order to cope with the environmental requirements, i.e. reducing GHG and reducing global warming, nuclear reactors are one of the top of candidates for providing a reliable energy source. Taking into account the considerable costs of pumping water for long distances, the need for a suitably sized nuclear reactor is clear. For islands, or remote areas, where the cost of transportation of fossil fuels (coal, oil or gas) is very expensive, nuclear reactors represent a favorable solution.

The shortage of water occurs mainly in developing countries with weak infrastructure and limited financial capabilities. These and other factors make small or medium nuclear power reactors an attractive candidate. Therefore, SMRs are perceived by several countries, as a convenient, economically competitive and viable source of energy which, when introduced, would not only complement the traditional energy sources, but would also promote technological development, serve as an incentive for social and economic progress, and secure the potable water needs.

8. CHALLENGES

Mankind is facing real challenges:

- Covering the shortage of fresh water, where about 1/4 of the world suffer from water problems,
- Reduction of energy poverty, where 34% of world population does not have access to commercial energy,
- Reduction of energy gap, and
- Reducing GHG and CO_2 emissions.

The only way to face the shortage of fresh water is the desalination of seawater, and the best energy source (electricity and/or heat) is a SMR nuclear reactor if we are really looking for reduction in the level of GHG and CO_2. This needs to:

- Build upon the real existing experience.
- Prove the safety
- Produce water at competitive costs, and
- Develop international co-operation in technological as well as investment and financing.

CONCLUSIONS

- Energy will continue to play a central role in global economic development, and fossil fuels have serious environmental impacts. Nuclear energy is the best known technology that is environmentally benign and reliable source of energy to generate electricity and heat. Its safety record compares favorably with alternatives.
- Desalination of seawater is the most promising solution to overcome the shortage of fresh water, and nuclear reactor in general, and SMR in particular, could be used safely.
- The experience gained shows that concerns over the use of nuclear energy in general and in nuclear desalination in particular have no grounds.
- International co-operation in the field of nuclear desalination is not only following the new trends all over the world for co-operation, but it also provides the best means to ensure safe and reliable application of nuclear energy in seawater desalination.

PANEL DISCUSSION
CHALLENGES OF SMALL AND MEDIUM REACTOR DEPLOYMENT

(Session 2)

Chairperson

J.B. Ritch III
World Nuclear Association

PROLIFERATION ASPECTS OF SMALL AND MEDIUM SIZED REACTORS

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Abstract

The paper deals with the proliferation risks associated with acquiring nuclear reactors and associated fuel cycle capabilities and gives the IAEA perspective on safeguards of nuclear power plants. It also discusses how the safeguards system might evolve if the number of reactors is going to increase dramatically – by a factor of ten or more. The responsibilities are incumbent upon nuclear suppliers, the States buying and using reactors and fuel cycle services, and upon IAEA safeguards. It is concluded that significant increases in nuclear power without proliferation are possible under arrangements that we can all accept.

1. INTRODUCTION

This Seminar has been convened with the aim of increasing the use of nuclear power, especially in States or regions having limited electricity distribution networks where small or medium sized reactors might be attractive. One of the concerns associated with nuclear power is the possibility that a State might acquire fissile material for use in nuclear weapons under the guise of a program established for peaceful purposes. If it is to fulfill its promise, then all of us gathered here, buyers, suppliers and verifiers, have to find the means to increase the use of nuclear power while preventing the spread of nuclear weapons.

In my remarks today, I will describe the proliferation risks associated with acquiring nuclear reactors and associated fuel cycle capabilities. Assuming that all of us here want to see the increased use of nuclear power, I will then describe what I believe are the responsibilities incumbent upon nuclear suppliers, the States buying and using reactors and fuel cycle services or capabilities, and upon IAEA safeguards. In relation to the means through which the IAEA might provide assurance against non-proliferation, I will share a few thoughts on how the safeguards system might evolve if the number of reactors is going to increase dramatically – by a factor of ten or more. I am convinced that we can have very significant increases in nuclear power without proliferation, and under arrangements that we can all accept.

2. PROLIFERATION RISKS

There are many ways that a State intent on acquiring nuclear weapons might proceed. For a State to acquire nuclear weapons through any means, it must have the will to do so, the capability to succeed, and be able to carry out its development and manufacturing activities without being detected. Fortunately for us, the world has changed from the early decades of the nuclear age when the restraints against proliferation were incomplete and ineffective. While the international non-proliferation regime has been evolving for nearly a half century, proliferation attempts discovered in the 1990s brought about a fundamental strengthening. As a result, controls on nuclear commerce are now more effective, proliferation detection capabilities are more sensitive, and the international community is more likely to act now with resolve to block any instance of proliferation, once detected.

The proliferation concern arises from the basic fact that regardless of the intentions of a State, peaceful nuclear power programs do create potential opportunities for States to acquire weapon-usable material. There are very substantial differences in the nature of those opportunities depending on the characteristics of the reactors and the arrangements pursued in relation to obtaining fresh fuel and for processing or disposing of spent fuel.

Every nuclear power reactor fueled with uranium produces plutonium as a natural byproduct – enough to make one or more nuclear weapons each year. Adding fuel cycle capabilities may provide possibilities for producing highly enriched uranium, for processing plutonium and uranium, and for concealing diversion or facility misuse. Since the adoption or expansion of a State's nuclear power program increases its "proliferation capability", every nuclear power program concerns the world community: is the program exclusively peaceful in character, or might it have a hidden purpose?

Each reactor, to differing extents, provides opportunities for diversion of the fresh fuel, diversion of spent fuel, and for undeclared production of plutonium or ^{233}U.

Adding additional reactors opens possibilities for higher plutonium production rates and additional possibilities for concealing diversion, e.g. through borrowing similar materials to avoid detection.

Each step of the fuel cycle mastered by a State opens possibilities for misuse of the facilities and for copying the technology in clandestine plants. Uranium enrichment plants and chemical reprocessing plants to extract plutonium from spent fuel are of greatest concern.

As the State's activities expand, so do its abilities to assist other States intent on acquiring nuclear weapons, whether unintentionally or in conspiracy.

3. PROLIFERATION-RESISTANCE AND SMRS

Are there specific features that would make SMRs less usable for nuclear weapons production and thereby avoid significant concerns amongst the international community? In the future, a new generation of nuclear reactors should emerge with "proliferation-resistant" features. Here are some steps that can be taken now to limit the extent to which nuclear power adds to the "proliferation capability" of a State:

a. Avoid the use of highly enriched uranium or plutonium in fresh fuel and consider plans for plutonium recycle only when the economic and waste management justifications are compelling;

b. Select reactors for high burn-up, which produce plutonium with low concentrations of ^{239}Pu (plutonium becomes more difficult to use in nuclear reactors as the percentages of other isotopes increase);

c. Select reactors that limit to the extent possible locations within or near to the reactor core into which fertile material could be introduced for clandestine production of plutonium or ^{233}U;

d. Incorporate features in the design of the reactor, and cooperated with the IAEA on inspection procedures and equipment to facilitate IAEA safeguards implementation.

Proliferation opportunities are not restricted to the reactors – in fact, the opportunities provided by fuel cycle operations may be of greater concern. Again, there are some steps that can be taken now to minimize the risk of proliferation associated with the fuel cycle:

a. Assure that reliable fresh fuel supply arrangements are made so as to remove or defer incentives for acquiring or developing uranium enrichment capabilities, and avoid stockpiling by arranging for the receipts of fresh fuel when required;
b. Use fuels that are difficult to process prior to irradiation and increasingly so upon irradiation; and
c. Adopt spent-fuel take-back arrangements, so as to remove or defer any incentive for the State to acquire or develop reprocessing capabilities.

Choosing the right options at the appropriate times makes the difference between contributing to solving future energy requirements in a responsible manner, and creating new worries for the international community.

4. THE BUYER'S OBLIGATIONS

If you, as a representative of a prospective buyer State, wish to send non-proliferation signals to your neighboring States and to the world community, here are a few thoughts to have in mind:

– If you define a national energy strategy based on realistic projections for energy demand and an analysis of alternative energy options, your needs for nuclear power will be easier for your citizens, your neighbors and the world to understand;
– Choose your reactors and fuel cycle arrangements from those offering the greatest proliferation resistance;
– Consider how you might establish confidence-building measures with your neighboring States, regional bodies, the IAEA and other international bodies (look into possibilities for obtaining the benefits of nuclear power as part of a multi-national arrangement for the ownership and operation of the reactors and for the provision of fresh fuel and the return of spent fuel);
– Take a positive attitude in relation to IAEA safeguards – check with States with established nuclear industries to determine how they have managed to make certain that the IAEA is able to carry out its role while you minimize the costs and interference that safeguards might otherwise bring.

5. THE SUPPLIER'S OBLIGATIONS

It is the responsibility of suppliers to offer facilities, equipment, materials and services that provide the least possible utility to a State if misused in a nuclear weapon program. Suppliers need to look at three principal concerns:

– Assure that the justification offered by a buyer State for the purchase of an SMR and any related fuel cycle service or capability is sound, appropriate and timely;
– Assure that the buyer State warrants the non-proliferation trust essential for nuclear commerce, including its:

a. acceptance and adherence to relevant non-proliferation Treaty obligations, especially under the NPT, together with applicable nuclear-weapon-free zone treaties; and

b. implementation of a comprehensive IAEA safeguards agreement and Additional Protocol;

– Conduct all commerce in conformance with nuclear supplier arrangements, including the Zangger Committee and the guidelines of the Nuclear Suppliers Group.

6. THE VERIFIER'S OBLIGATIONS

IAEA safeguards are considered by many to be the "cornerstone" of the non-proliferation regime. Successful IAEA verification requires that:

a. The buyer State has a safeguards agreement (preferably with an Additional Protocol) and subsidiary arrangements in force, and a competent State System of Accounting for and Control of Nuclear Material (SSAC) to carry out the national obligations required under the safeguards agreement;

b. Adequate funding for the IAEA for inspectors and inspection equipment;

c. The IAEA obtains the design information and other information required from the State, and examines and verifies the design information on facilities declared by the State, initially, when safeguards are first applied at a facility, and periodically, over the life-cycle of each facility;

d. IAEA inspections are carried out at declared facilities according to IAEA technical criteria, including complementary access activities as deemed necessary by the IAEA; and

e. Discrepancies and anomalies are resolved promptly and conclusively.

7. SUPPOSE THAT NUCLEAR POWER REALLY TAKES OFF

There are 262 designated IAEA inspectors now, and combining the regular budget with extrabudgetary contributions, the total annual cost of IAEA safeguards now is just over $90M. If there is to be a significant expansion in nuclear energy, say a ten-fold increase, it would be very difficult to get a corresponding increase in the IAEA budget. If nuclear power is to increase, we will need to find ways to continue to provide credible and independent assurance against proliferation. There are moves underway now that are expected to reduce the current safeguards costs in States with comprehensive safeguards agreements, including the Additional Protocol. These savings are based upon the expectation that the assurance provided against undeclared nuclear materials and operations within a State obtained through the implementation of the Additional Protocol would allow some of the traditional measures to be relaxed or even discontinued. Based upon current expectations, however, those savings will be limited and significant increases in IAEA funding would still be required if nuclear power were to take off.

Enhancing the "proliferation resistance" of future reactors and fuel cycles should reduce the verification burden necessary. If the innovative reactors and fuel cycles now coming under consideration make proliferation much more difficult to carry out, and make verification easier and less costly, then the funding required for "effective" safeguards might be reduced. It is too early for specific conclusions; certainly there are interesting concepts on the table, but it is too soon to specify the principles that should apply, the rules that should be followed and alternative verification systems and arrangements that might emerge. The promise lies, I believe, in three complementary elements of what may constitute the future non-proliferation system. Those are:

1. The technical concepts themselves, and how they might make it possible for user States to derive the benefits of stable and reliable electricity supplies with very limited physical possibilities for diverting or producing nuclear materials that could be used to fashion nuclear weapons or any nuclear explosive device. Under the U.S. Generation IV program and the Gen IV International Forum, promising new reactor designs will be selected for development and future commercialization. Under the IAEA INPRO Program, efforts are being made to select a new generation of nuclear reactors that will be economical, incorporate inherent safety principles, be proliferation-resistant and generate minimum waste.
2. The institutional framework for non-proliferation, which will continue to rest upon the NPT and comprehensive IAEA safeguards agreements (with the Additional Protocol), but may grow to include multi-national energy parks, all-in, all-out fuel supply arrangements, and possibly a new convention on the peaceful use of nuclear energy with obligations on suppliers, buyers, international controls on nuclear commerce and verification by the IAEA.
3. IAEA verification itself, in which the requirements are chosen in consideration of the proliferation-resistant features of the reactors and fuel cycle arrangements, taking into account the State's existing and future nuclear capabilities and the extent to which it honors its non-proliferation undertakings. New verification methods will almost certainly reflect the information age, with installed, authenticated sensors integrated into reactors and fuel cycle operations, secure communications via internet, and automated intelligent review of such verification data.

CONCLUDING REMARKS

Even now it is clear that expanding nuclear power with no further proliferation will require a compact between buyers, suppliers and verifiers in which the obligations of each party are clearly specified. Even now there are obligations and opportunities for supplier States and user States. Buyers wishing to demonstrate their commitment to non-proliferation should plan their reactor and fuel cycle services appropriately and carry out their programs in a transparent manner. Suppliers have a special responsibility to ensure that their wares are not acquired under false pretenses, or that once supplied, they are not diverted to weapons production. This holds true not only for research and power reactors, but especially for fuel cycle technology transfers.

Nuclear power provides a means to bring peoples together. Stable and safe electricity supplies serve as a constructive means not only for nations to improve their economies, but also to improve regional relations.

The world needs nuclear power for peaceful purposes, and concrete steps towards the phased elimination of existing nuclear arsenals. While no State has produced nuclear weapons solely based on its civilian nuclear power program, we all, buyers, suppliers and verifiers, must base our beliefs and actions on the possibility that if a State were to pursue development of nuclear weapons, it is in principle possible to misuse a peaceful nuclear program for that purpose. The world must remain vigilant that this does not happen. We, as buyers, suppliers and verifiers, share a unique responsibility for the future, through proliferation-free nuclear power.

IAEA-SR-218/88

INFRASTRUCTURE REQUIREMENTS FOR SUCCESSFUL DEPLOYMENT OF NUCLEAR POWER

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Abstract

The viability of a nuclear power programme in any country depends on the availability of adequate infrastructure. The infrastructure requirements include legal, institutional, educational, technological, industrial, financial and human resources. In principle, any country can develop its infrastructures to an adequate level, but this requires substantial time and effort. Small and medium reactors and innovative reactors that could possibly minimize the infrastructure requirements would offer an attractive option to developing countries unable to invest in upgrading the infrastructure in a big way. This paper discusses the infrastructure requirements for the deployment of nuclear power.

1. INTRODUCTION

A reliable and adequate supply of energy, and especially of electricity, is indispensable for economic development. Thus providing safe, reliable energy in economically acceptable ways is an essential political, economic and social requirement. Planning and decision making for energy and electricity supply are important for governments. National governments will probably have laid down policies in such sectors as national development (including goals and priorities), energy development (including supply) and international relations. These policies would be of a long-term nature and where they are the result of consensus, they would not be expected to change with political changes in the country.

Each country will decide on the level and extent of national participation desired at each stage in its nuclear power programme. However, it must be emphasized that there is a minimum level necessary. First, the future owner organization must be well informed and the regulatory authority must know what its responsibilities will be. This means that there must exist a group of well-qualified, well-trained staff, with experience which they will have acquired most often from abroad. Secondly, a country must be able to accept the responsibility to reach the minimum level of national participation to achieve an acceptable and assured level of safety as well as to make nuclear power a viable energy option. The desirable level of participation must be seen against the existing infrastructures in the country and the levels to which it is possible and appropriate to develop these. In this context infrastructures have been defined as: organizational and regulatory frameworks; qualified personnel, and education and training capabilities for acquiring such personnel; financial capabilities; industrial capabilities; and R&D capabilities. A common approach has been that the first plant is ordered under a turnkey contract and steady progress is then made with subsequent plant orders towards split package and multiple package contracts, each step placing increasing demands on the domestic infrastructures.

The purpose of this paper is to highlight areas where such policy decisions are needed, the options available and the contexts in which they should be considered.

2. LEGAL AND REGULATORY INFRASTRUCTURE

Responsibility for development of the infrastructure to create, regulate and maintain a nuclear power program rests with the government, national organizations and institutions. Establishment of a nuclear power program entails legal requirements at both the national and international level.

2.1. Nuclear Safety Regulatory Authority

The government must establish a system to develop nuclear safety regulations, issue operating licenses and perform inspections so regulations are met and standards followed. Legislation must be enacted to create and empower a nuclear safety regulatory authority. This regulatory authority must be independent of the operator and have the legal power to:

- Formulate rules and regulations to be followed by the owner/operator;
- Issue licenses or permits for siting, construction, commissioning, operation and decommissioning of nuclear power plants;
- Supervise measures ensuring that rules and regulations are followed by owner/operators;
- Ensure that the licensee understands its obligations and is competent to fulfill them;
- Enforce laws.

The Convention on Nuclear Safety stipulates that other parties to the Convention in the vicinity of a proposed installation be given enough information to enable them to make their own assessment of the likely safety impact on their own territory.

To provide support at the international level, the IAEA has published fundamental safety concepts as well as Codes and Safety Guides as part of its Nuclear Safety Standards (NUSS) programme. It is important that only one organization, the owner/operator, has primary responsibility for the safety of a plant. As a prerequisite for obtaining an operating license the owner/operator must accept this responsibility, which cannot be shared either with the plant designer or constructor or with the authority which regulates safety in the country.

An IAEA International Regulatory Review Team (IRRT) can review the adequacy of regulatory authorities upon government request. The Convention on Nuclear Safety foresees that signatory States will report on measures taken to maintain a high level of safety and that these reports will be discussed in periodic review meetings.

2.1.1 Defense in Depth

A publication of the International Nuclear Safety Advisory Group (INSAG) entitled *Basic Safety Principles for Nuclear Power Plants* discusses the need for a defense in depth concept centered on several levels of protection, including successive barriers to prevent the release of radioactive materials to the environment. The objectives are:

- To compensate for potential human and component failures,
- To maintain the effectiveness of the barriers by averting damage to the plant and to the barriers themselves,

- To protect the public and the environment from harm in the event that these barriers are not fully effective.

INSAG has further developed requirements for a defense in depth strategy in a more recent publication. In this strategy, accident prevention is the first priority. However, if preventive measures fail, mitigating measures, in particular a well designed confinement system, can provide additional protection for the public and the environment.

2.1.2 Quality Assurance

It is important to achieve the highest levels of quality in all stages of a nuclear power project, from site selection through design, construction and commissioning to operation and decommissioning. This is indicated by the fact that quality assurance (QA) is one of the five main topics of the Codes and Safety Guides issued in the IAEA's NUSS program. Quality assurance is defined as: "all those planned and systematic actions necessary to provide adequate confidence that an item or service will satisfy given requirements for quality".

The recently revised NUSS Code and Safety Guides on QA put greater emphasis on the responsibility of everyone concerned to achieve their performance objectives.

1. Management is responsible and accountable for all aspects of quality of performance, including planning, organization, direction, control and support

2. The line unit is responsible and accountable for achieving quality of performance to ensure safety and reliability.

3. The assessment unit evaluates the effectiveness of the management and line units in carrying out their responsibilities to achieve quality of performance, and identifies and ensures removal of barriers which may hinder the ability of the plant organization to function effectively in carrying out its responsibilities.

2.1.3 Safety Culture

Safety culture is a concept, which can be described as inculcating in all personnel a pervasive safety consciousness, a commitment to excellence and personal accountability. A safety culture should be established in all countries which operate nuclear power plants and codified in laws, regulations and standards for nuclear safety. Many IAEA Member States have already shown their commitment to this idea by consenting to be bound by the Convention on Nuclear Safety, which entered into force in October 1996.

Responsibility for the safety of nuclear installations and radiation protection must be defined by law, as must the responsibility of the plant operator and the regulatory authority or authorities (where radiation protection and nuclear safety regulatory bodies are separate).

2.2. Radiation Protection Regulatory Authority

A national system for radiation protection is a precondition for nuclear activities in a country. If this does not exist, the first step is for the government to enact legislation and empower a regulatory authority and establish regulations and standards for radiation protection. The regulatory authority licenses users of radioactive materials and radiation sources and ensures that regulations are followed. The Convention on Nuclear Safety addresses these issues. The adequacy of the system can be checked by an IAEA Radiation Protection Advisory Team (RAPAT) on government request.

Basic Safety Standards published by the IAEA are the only international standards available in the area of radiation safety. Therefore, many countries accept the BSS as national standards. The IAEA has a statutory right and obligation to require that the BSS be used in all projects it supports in a particular country. A number of countries have their own standards which differ from the BSS in some respects.

Experience from well managed nuclear power plants shows that occupational and public exposures were kept to a fraction of the annual dose limits stipulated by safety standards .

Within the radiation protection regime a policy should be defined for managing radiation emergencies. This is needed not only for nuclear power plants but also for accidents with radiation sources, which can have considerable local impact.

2.3. Third Party Liability

Liability for nuclear damage is part of the legal framework that has developed around the peaceful uses of nuclear energy. The present international liability regime is embodied primarily in two instruments: the Vienna Convention on Civil Liability for Nuclear Damage (1963) and the Paris Convention on Third Party Liability in the Field of Nuclear Energy (1960). These are linked by a Joint Protocol adopted in 1988. The Paris Convention was later extended by the 1963 Brussels Supplementary Convention. These Conventions are based on concepts of civil law and share the following main principles:

- The international liability regime applies to nuclear installations defined in the Conventions, e.g. civil, land based nuclear reactors and reprocessing and storage facilities, as well as nuclear materials transported to or from such installations.
- Liability is channeled exclusively to the operator of the nuclear installation.
- Liability of the operator is absolute, i.e. the operator is held liable irrespective of fault.
- Liability is limited in amount.
- Liability is limited in time.
- There will be no discrimination of victims on the grounds of nationality, domicile or residence.

Following several years of preparation, a diplomatic conference held at the IAEA Headquarters in September 1997 adopted a protocol to amend the Vienna Convention and a Convention on Supplementary Compensation for Nuclear Damage.

2.4. Non-Proliferation Regime

Since the first international transfer of nuclear fuel, equipment and technology, assurances of exclusively peaceful use have generally been a condition for supplies under bilateral agreements between a recipient and a supplier State. These agreements generally permitted verification by the authorities of the supplier State. Since the early 1960s, this verification of specific supplies has been in most cases delegated to the IAEA through its safeguards system, a function which had been foreseen in its Statute.

Subsequently, an international non-proliferation regime came into existence. The basis of this regime is the Treaty on the Non-Proliferation of Nuclear Weapons (Non-Proliferation Treaty, NPT), which entered into force in 1970. Any State (with the exception of the five proclaimed nuclear weapon States, China, France, Russia, UK and the USA) which becomes a party to the NPT makes the commitment not to receive, manufacture or otherwise acquire nuclear weapons or other nuclear explosive devices, and to accept IAEA safeguards on all of its nuclear materials in all of its current and future peaceful nuclear activities (known as full scope or comprehensive safeguards). A conference held in 1995 reviewed the operation of the Treaty and decided on its indefinite extension.

Supplier States started to discuss in various forums (e.g. the Zangger Committee and the Nuclear Suppliers Group, NSG) common conditions for supplies during the late 1970s. States participating in the NSG have agreed that a condition for nuclear supplies will be acceptance of full scope safeguards under the terms of international agreements such as the NPT, the Treaty for the Prohibition of Nuclear Weapons in Latin America (Tlatelolco Treaty), the South Pacific Nuclear Free Zone Treaty (Rarotonga Treaty), the African Nuclear-Weapon-Free Zone Treaty (Pelindaba Treaty) or the Southeast Asia Nuclear-Weapon-Free Zone Treaty (Bangkok Treaty). Earlier, specific supplies could be obtained under a safeguards agreement which covered only the supplies in question but this is no longer possible from any of the NSG countries. In some cases a bilateral agreement between the supplier State and the purchasing country is also required.

2.5. Environmental Regulations

The increasing use of energy worldwide has become a major environmental concern since energy use has environmental impacts at all levels:

- Locally, e.g. through use of primitive cooking stoves in many developing countries, smog formation in urban areas, and local flooding and resettlement as a result of new hydropower schemes;
- Regionally, through acid rain caused by emissions of sulphur dioxide and nitrogen oxides;
- Globally, through the contributions of carbon dioxide and methane to the atmosphere.

The greenhouse effect and global warming now seem to be a major subject for discussion. Emissions of sulphur dioxide and nitrogen oxides from fossil fuelled power plants can be limited by flue gas cleaning, though at a cost; carbon dioxide emissions are only limited by reducing fossil fuel use, which will influence electricity supply systems. Regardless of international environmental goals, all countries must protect the environment in their national energy policies by reduction or at least control of emissions. Nuclear power can contribute in this context as emissions from normal operation are very small.

2.6. Public Acceptance and Participation in Decision Making

Public acceptance is a very important issue for nuclear power. Attitudes vary from country to country. In some countries there is acceptance of nuclear power. In other countries, both industrialized and developing, public opinion has turned against nuclear power and this is often cited as a major obstacle to its further development. The arguments used against nuclear power focus on three issues:

- The risk of repetition of a serious reactor accident with consequences like those of the Chernobyl accident,
- The claim that the waste presents a problem that has no solution,
- The alleged close link between civilian nuclear power and nuclear weapons.

There should be no doubt that these arguments have caused fear among the public but, at the same time, it appears that very often the public has been neither well informed nor directly concerned, with side issues sometimes dominating the debate. Experience has shown that the only way to influence public opinion is through a carefully designed long-term education programme based on correct, neutral information. Such a programme requires a major effort but its importance should not be underestimated.

With industrial development, governments and parliaments became the guardians of public safety and took the decisions needed to establish new plants and carry through programmes. This led to the creation of local consultation procedures which were to be carried out before decisions could be taken on the siting of new and potentially hazardous industries. Under all circumstances it is important that there be a process of local consultation and that it be accessible and transparent.

At the local level the role of politicians in public participation has often been very useful. At this level they have more direct contact with their electorate, see the importance of local issues and can serve as a channel for information to their constituency. This has led some countries (e.g. France, Hungary and Sweden) to establish local information or safety committees which have direct insight into the safety, operation and emergency planning at a plant.

3. INSTITUTIONAL INFRASTRUCTURE

3.1. National Energy Policy Development

A country considering a nuclear power programme would have a national energy plan specifying the objectives for the national energy policy. The objectives include:

- Improved energy independence
- Development of indigenous energy resources
- Economic optimization of energy and electricity supply
- Stability of electric grid system
- Availability of energy at prices which support general development
- Environmental protection
- Opening of competition in electricity market.

Some of the above objectives are, of course, overlapping and may yield the same energy policy. Some of the policy options could preclude the use of nuclear power in a country. For example, if a primary objective is to use indigenous energy sources this would not favour the introduction of nuclear power plants. It would be necessary for a nuclear power programme to have a well defined role within the overall energy policy.

3.2. Energy and electricity planning

In most countries with nuclear power plants, base loaded nuclear electricity competes favorably with other options, particularly coal and oil. With recent developments in many countries to introduce competition into electricity markets, the costs of fuel and of operation and maintenance for nuclear power plants have been successfully lowered to remain competitive, without jeopardizing plant safety.

In the overall context, there are economic benefits from a nuclear power programme going beyond the mere comparison of electricity costs between alternatives. An important consideration in many developing countries has been the positive influence of a nuclear programme on the technological sophistication of the country. On the other hand, certain additional costs are directly related to the introduction of nuclear power, such as the cost of establishing a regulatory infrastructure. It would be desirable if these costs could be distributed over a number of plants, leading to the conclusion that a nuclear programme must be large enough to enable spreading of costs to yield economies of scale.

3.3. Nuclear Fuel Cycle Management

The nuclear fuel cycle consists of a number of distinct industrial activities which can be separated into two sections: the front end, comprising those steps prior to fuel irradiation in the plant; and the back end, including the activities concerning the irradiated, spent fuel.

3.3.1 Front End

Acquisition of its first nuclear power plant by a country involves a major degree of dependence on external suppliers, with associated commitments to non-proliferation and international cooperation. The power plant is usually provided with fuel for one to four years of operation but it must be re-supplied over its lifetime of 40 or more years. When the type of power plant is decided, the choice of the form of the fuel is made:

A desire to assure fuel supplies over the lifetime of a reactor (40 years or more) leads to considering establishing a domestic fuel supply and fuel production technology to guarantee continual operation of the plant. With the exception of enrichment, front end technologies are available for transfer, usually on commercial terms through licensing. The counter-argument to domestic front end fuel services is that at present it is hardly economic. It is normally cheaper and as reliable to use the international market for fuel supplies.

Commercial enrichment services are available to any prospective buyer with good non-proliferation standing. Services to convert uranium to chemical forms required for enrichment are also widely available and competitively priced. Thus, in all aspects of the front end of the fuel cycle, security of supply is not a serious concern.

3.3.2. Back End

In the back end of the fuel cycle there are three policy options for management of the spent fuel:

- Reprocessing for fabrication of mixed oxide fuels (MOX) fuel to be recycled in light water reactors (LWR),
- Storage for 30–50 years and subsequent disposal as high level waste (HLW) (the once-through cycle),
- Deferral of the decision on whether to reprocess or dispose of the spent fuel.

Reprocessing is now offered by three countries, but at least two (France and UK) require that the resulting HLW be returned to the client country with the separated uranium and plutonium. Thus, plans must be made for domestic HLW disposal, whichever back end option is chosen. Experience has shown that international transport and storage of both plutonium and vitrified HLW can be highly problematic as they have become focal points for public and international opposition, even though a high level of safety can be ensured.

The second option of storage and final disposal of the spent fuel without reprocessing is chosen by many countries at present (e.g. Germany, Sweden and USA) and HLW disposal technology is being developed to meet future requirements. In Canada, the decision not to reprocess fuel from its CANDU type PHWRs was taken long ago. Power plants in some countries were designed for ten years of spent fuel storage, with extra storage added later. This has sometimes been provided through lower cost dry storage facilities.

The third option, chosen by many countries, of deferring the back end decision is the cheapest as it permits deferral of decisions on HLW disposal and siting. However, it could be an easy opening for attack by those in opposition maintaining that there is an unsolved waste problem.

3.4. Managing Radioactive Waste and Decommissioning Nuclear Facilities

3.4.1 Waste Management and Disposal

Radioactive waste has become a focus of environmental concerns related to nuclear power. The main feature of wastes from nuclear power plants is that they occur in small quantities, and can therefore be more easily managed and disposed of. Radioactive wastes are divided into three categories:

- Low level waste (LLW) arises from nuclear plants and from applications of radioisotopes in medicine, industry and research, and must be isolated for a periods of up to about 200 years.
- Intermediate level waste (ILW) consists to a great extent of operational wastes from power plants, such as ion exchange resins, and can usually be treated and disposed of in the same general manner as LLW.
- High level waste (HLW) consists of fission products and plutonium contained in spent fuel elements and must be safely isolated from the environment for very long periods, possibly hundreds of thousands of years. HLW also generates heat, which can be significant for the first 30–50 years.

Safe waste management involves the application of technology and resources to limit the exposure of the public and workers to ionizing radiation and to protect the environment from radioactive releases, in accordance with national regulations and international standards.

Further international progress was made in this area with the adoption of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.

To assist national regulatory bodies, the IAEA is revising its Safety Series publications to be issued within the Radioactive Waste Safety Standards (RADWASS) program so that the structure is equivalent to that of the NUSS programme.

In each country where radioactive materials are handled, a national waste management program must be established. It should also ensure continuing communication between the regulatory authorities, the operators and the public.

3.4.2 Decommissioning Nuclear Facilities

A nuclear power plant must be decommissioned at the end of its useful life. A useful life of 30 years is often referred to but plants are usually designed for 40 years of operation. This lifetime can be extended beyond 40 years with suitable management including control of degradation processes, maintenance, repair and refurbishing and/or replacement of plant components and systems. There are essentially two options for decommissioning a plant:

- The plant is dismantled after operation ceases and the site is restored or adapted for reuse.
- Fuel is discharged to a storage facility and non-radioactive components are dismantled but radioactive parts are mothballed for 30-50 years or longer before dismantling.

The first option has the benefit of freeing potentially valuable sites for other purposes, notably for new power plants, as early as possible. It also neutralizes continuing public concern about whether the reactor remains a threat to public health and safety.

The second option has the benefit of reducing the total radiation dose to decommissioning workers as radioactivity will have decayed substantially in the 30–50 year mothball period. This also reduces the cost of dismantling, though the saving may be offset by the cost of maintenance and surveillance during the mothballing period. Technology is available for dismantling radioactive reactors but new technology may be developed over the next 30–50 years to allow further reduction of costs and worker exposure. In both cases, some radioactive materials will have to be managed as waste as a result of dismantling. Three prerequisites must be satisfied to decommission a nuclear power plant

- Well trained personnel with appropriate technical skills,
- A licensed storage or disposal facility to accommodate decommissioning wastes,
- A regulatory basis for implementing a decommissioning project.

The IAEA has published decommissioning guidelines for research reactors and small facilities but they can, to a great extent, also apply to large facilities. There is now a need for more specific guidance on the development of decommissioning regulations. IAEA safety standards on decommissioning are therefore being developed as part of the RADWASS program.

3.5. Public Information and Education

In many countries nuclear power is encountering strong public opposition. Gaining public acceptance will require informing and educating the public correctly and neutrally. Therefore, a carefully planned information and education strategy would need to be formulated and implemented at an early stage, on the basis of an understanding of the level of public knowledge and of the public concerns.

Local benefits will accrue from the introduction of a large industrial plant and are likely to increase local support for such a project. Benefits may include added employment opportunities, improved education possibilities and greater local commerce.

4. TRANSPORT INFRASTRUCTURE

4.1. Electric Grid

The current nuclear power plants were developed for use in large interconnected electric grid systems of high quality and hence are likely to face problems when introduced into small or weaker grids. The major causes of poor grid voltage and frequency stability include insufficient generation and interconnection; inadequate control equipment and load dispatching; unreliability of protective systems; and non-optimal grid operation management and lack of coordination between different generating stations. Thus, limited amount of generating capacity and fragmented transmission grids are the most important factors limiting or delaying the possible introduction of nuclear power in a county. However, this situation could improve if small and medium sized reactors are available in the market.

4.2. Road, Rail and Water Transport

When a major project such as a nuclear power plant is implemented, improvements or extensions in the transportation sector may be essential. New roads, railways or ports with heavier load carrying capability may be necessary for delivery of equipment and material. For each mode of transportation, an assessment should be made regarding the suitability of routes for transporting equipment and components from foreign suppliers and domestic manufacturers to the project site. Bridges and culverts may have to be strengthened and obstructions cleared. In most cases, even with a well developed transportation sector, a new project site for a nuclear plant may require some extension of the existing facilities. This may be a major project in itself with its own impact on investment and debt but there would also be positive spin-off benefits to the country's transport infrastructure.

5. COMMUNICATIONS

The communications network is the neurological system of a nation. This system links sources of data such as trade and industry to organizations using information to make decisions both in the government and the private sector. The advances in information technology and communications provide invaluable tools to improve the productivity and efficiency in the electricity industry as a whole. The role of good communications support pervades all aspects of nuclear power utilization.

6. FINANCIAL INFRASTRUCTURE

6.1. Role of Government

The commitment of the government to a nuclear power programme, together with strong policy support, is of paramount importance in order to reduce the uncertainties and associated risks and improve the overall climate for financing. The government should prepare long term plans for nuclear power development, clearly describing the role of nuclear power in the national energy plan, as well as the associated financial and economic plans. The government should also ensure that the necessary infrastructure is developed to support the introduction of nuclear power. A regulatory system for licensing nuclear power plants must be in place.

The investment climate is improved if the government and the owner/operator achieve good records of consistent and fair dealing with lenders and investors. Only countries with acceptable credit ratings would qualify for bank loans and other credits for financing a nuclear power project. The development of sound economic policies as well as good debt management and appropriate sharing of project risks would all contribute to this end.

6.2. Key Criteria

For successfully financing a nuclear power project in a developing country, it is essential for the government as well as the utility to do the following:

- Commit itself to the nuclear power programme.
- Make a thorough financial analysis together with an economic analysis for evaluating the feasibility of the project.
- Ensure that the construction programme is well planned and regulatory issues are fully addressed before construction starts in order to minimize the risk of expensive delays.
- Maintain generally acceptable credit ratings in order to obtain investments and debt financing.
- Finance as much as possible of the local cost component of the project in local currency from sources within the host country itself. The importance and complexity of this are often underestimated.
- Set electricity tariffs at a level necessary for a sound financial position.
- Build up strong management capabilities and utilize thoroughly a full range of expertise to deal with the financial complexities.

The potential benefits of nuclear power include a certain buffering against escalating fossil fuel prices, which helps maintain the long term stability of electricity prices. However, because of the higher capital costs of nuclear power plants, the financing of a nuclear program is sensitive to inflation. Financing schemes and the related issue of supply contracts are therefore essential considerations. Three types of supply contracts have commonly been used in the past: turnkey, split package and multiple package. In recent years, two new supply mechanisms have been used for fossil fuelled power plants: build–own–operate (BOO) and build–operate–transfer (BOT).

7. HUMAN RESOURCE DEVELOPMENT INFRASTRUCTURE

The technological, safety and reliability requirements of a nuclear power programme dictate the careful selection and recruitment of highly qualified and competent personnel by plant owners as well as by regulatory organizations. This can prove to be a national asset and also give an impetus for raising the level of national technical education and training capabilities, which will be beneficial for other industries. Moreover, human resource development infrastructure is a prerequisite for successful technology transfer. Human resource development requires long lead times and this aspect is not frequently taken into account in programme planning. Evaluation and upgrading, if necessary, of the education and training systems available in the country should be a priority for embarking on a nuclear power programme. The system should cover university education, vocational training and specialized training.

8. INDUSTRIAL INFRASTRUCTURE

There are no firm requirements regarding the industrial support infrastructure for starting a nuclear power programme. However, the plants have to be built, the equipment and components have to be installed and tested, and the plants have to be operated and maintained within the country. This translates into a requirement at some stage in the programme, availability of industrial support infrastructure for material, components and services. The capabilities could be acquired through technology transfer from the vendor countries. Experience shows that until now, no country with a very low level of industrial infrastructure has successfully incorporated nuclear power. This, in itself, is not a sufficient reason to affirm that such countries cannot or should not go for nuclear power.

9. REGIONAL COOPERATION

Policy concerning relations with neighbouring countries within a region is increasing in importance, as shown by the number of regional associations and alliances being formed for various purposes. This applies also in the case of nuclear power programmes as there are many areas, including the following, in which regional co-operation could yield direct benefits:

- Electric grid integration
- Nuclear safety
- Environmental protection
- Sharing of plant services
- General R&D and human resources development
- Nuclear fuel cycle
- Non-proliferation assurances

It is not necessary that all parties to a regional co-operation agreement share an interest in nuclear power and its development. For example, while Sweden and Finland have important nuclear power programmes, Denmark is opposed to nuclear power. This has not prevented good and rewarding co-operation on nuclear safety matters.

CONCLUSION

Several areas related to infrastructure requirements have been highlighted in the above discussion, where policy decisions are needed and available options along with their implications need to be considered prior to establishing a nuclear power programme. The IAEA could provide technical assistance and expert services to requesting member states in planning and implementing their nuclear power programmes as well as in strengthening nuclear power infrastructure.

IAEA-SR-218/41

CURRENT STATUS AND PROSPECTS OF SMALL NUCLEAR POWER REACTORS IN THE RUSSIAN FEDERATION

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Abstract

A considerable part of the Russian Federationn territory falls in the regions of North and Far East, which occupy more than 60% of its total area. The consumers' power supply is primarily decentralized and some local demands for electric and thermal energy do not exceed 100–150 MW. Thus in the Russian Federation there is a large market for power sources including small nuclear power plants (SNPP). The fundamental reorganization of social and political system as well as restructuring of the economy in the country require the revision of those plans and intentions of constructing SNPPs that were set previously. At the same time both in the past and today, it is clear that in the XXIst Century the unique fossil and regenerative natural resources of northern and eastern regions of the Russian Federation will be in intensive demand. Therefore the principal targets and prerequisites for developing small nuclear power in our country remain the same.

1. INTRODUCTION

The very notion of "small power" as applied to nuclear reactors and/or nuclear plants has undergone significant changes in the numerical definition with the progress in developing nuclear power in the world.

The lower power limit for SNPPs has never been discussed with a view to its specific numerical value. Actually it may be any value if, proceeding from the particular objectives and conditions of plant construction and operation, all the intellectual, financial, material and other expenses and costs of power supply implementation are justified.

The upper power limit for SNPPs was always rather vague and varied (increased) with the nuclear power development. So in the 50s–60s the power reactors with the capacity of 12–25 MW, sometimes up to 50 MW, used to be assigned to the reactors of small power. Today the IAEA proposes that reactors with the electric power up to 300 MW should be considered as "small" reactors.

In the Russian Federation both in the past and today the notion of "small" electric and heat generating units are almost adequately related to the demand of consumers located on the vast territory of decentralized power supply. This is more than 60 % of the total area of the Russian Federation, primarily the regions of North and Far East. So, naturally taking into account the IAEA approaches, we de facto are guided by the results of studying the current state and potential needs for energy in these country regions, which actually form the potential market for the units of small (non–nuclear and nuclear) power sources. Practically it means that as a rule the nuclear plants with reactors of thermal power not exceeding 150–200 MW are referred to "small" power plants.

The problem of power supply to the isolated regions of North and Far East of the Russian Federation is traditionally solved by constructing electric and thermal energy sources with the aim to provide power supply to individual, widely spaced industrial facilities, large and small populated areas, sea ports and other consumers. It results from the so–called "spotty", i.e. not compact and selective pattern of developing these territories. Thus local, disconnected from each other (isolated), self-sustained power grids are being formed and operated. Some of them are fairly large (several hundreds MW(e)), such as the Norilsk, Western Yakutsk, Central-Yakutsk, Central Magadan power grids. However for the most part these are relatively small (several tens of MW(e)) and very small power grids. In addition there are more than twelve thousand very small (less than 1000 kW(e)) independent diesel power plants and a significantly higher number of small boiler plants in operation in these regions.

The number of power plants incorporated into local power grids is not high, only about one hundred. However, they generate nearly 90% of electric power being consumed in the area of North. The capacities of these power plants essentially cover the range from 10 to 100 MW(e). Hence it becomes quite obvious, that if this vast region were considered in the context of possible application of nuclear power sources, it would be quite natural first of all to find out how feasible and reasonable it will be to incorporate them into the local power grids. In this case it is significant that generally the tasks of power and heat supply should be closely linked.

2. BACKGROUND [1], [2]

The systematic work in the field of small nuclear power engineering was initiated in the USSR in the mid–1950s, at the time when the world's first NPP was commissioned in Obninsk. The driving force for its onset and further development was the recognition that the tremendous energy content of nuclear fuel can give decisive advantages of using nuclear power sources on the great area of Northern the Russian Federation. In these areas large amounts of conventional fossil fuel had to be transported over large distances to the places of its consumption. In some Arctic regions the transport share in the cost of fuel delivered to the site is equal to 80–90 % of the total cost.

In the period before 1965 the following pilot and demonstration NPPs of small power were constructed and put into operation: TES–3 in Obninsk; ARBUS and VK–50 in Dimitrovgrad (Table I.).

TABLE I. SMALL NUCLEAR POWER PLANTS CONSTRUCTED IN THE USSR

Name	Reactor type	Reactor thermal power, MW(t)	Capacity, e/t, MW	Number of units	Location and the year of commissioning
TES-3	PWR	11	1.5/-	1	Obninsk, 1961
ARBUS	Power reactor with organic moderator and coolant	4	0.75/-	1	Dimitrovgrad, 1963
VK-50	BWR	250	50/-	1	Dimitrovgrad, 1965
Bilibino nuclear power plant (BiNPP) (co-generation)	Channel-type water-graphite	62	12/17,5-29	4	Bilibino, 1974-1976

In 1963 the work was initiated with the aim to choose the site and to design the first commercial SNPP. That was the Bilibino NCGP, constructed in the settlement of Bilibino in one of the most remote, hard–to–reach fuel–deficient regions in the Far North–East of the country, in Chukotka. In 1974–76 all the four power units of this SNPP were put consecutively into operation and connected to the local Chaun–Bilibino power grid.

The Bilibino NPP drastically improved power supply of the entire Chaun–Bilibino industrial region. Actually it generates the main share of electricity and meets almost all the demands (95,5%) of Bilibino for district heating. Moreover, in the course of its operation the BiNPP far exceeded its rated parameters of electricity and heat production, generating up to 352 million kWh of electricity and up to 312 thousand Gcal of thermal energy per year (the design performance characteristics are equal to 278,3 million KWh and 270 thousand Gcal per year, respectively).

With the highest economic efficiency and the lowest fuel cost, the Bilibino NPP covers the basic and, to a large extent, peak loads, while operating according to the electricity consumption schedule. Daily schedules of electric loads are very non-uniform, due to that the BiNPP has to operate in the mode of frequency control in the grid. The BiNPP has high load following capabilities which make it possible to vary the electric power within the range from 100% to 50% with the rate of 0,4% N/s (N is the power level at the given moment). During the heating period the BiNPP follows the heat load schedule likewise any co-generation plant.

Being part of the Chaun–Bilibino power grid together with the organic fuel fired power plants, the BiNPP had the load factor within the range from 0,702 to 0,837 every year up to 1993 (the number of effective full power hours was from 6149 to 7333 per year).

During ten years of its operation, from 1985 to 1994, the BNCGP made it possible to save more than 2.1 million tons of fossil fuel or about 3 million tons in terms of local coal, i.e. ~ 300 thousand tons of coal annually.

During the last ten years, when the level of power consumption by industrial facilities decreased drastically and the northern power grids introduced certain control limits, the BiNPP operating parameters have significantly decreased (the load factor dropped to 0,35–0,40). However, in spite of that the BiNPP was and still is the central link in the Chaun–Bilibino power grid.

The construction and successful operation of the BiNPP is the key stage in developing small nuclear power engineering in the Russian Federation. It testified to the real implementation and economically efficient use of SNPPs under the extreme conditions of northern regions. A separate paper specifically dedicated to this very important issue is presented at this seminar.

During the 60s, 70s and the first half of 80s numerous SNPP designs were studied, primarily with water–cooled (PWR and water–graphite) reactors. Two expeditions were carried out in the regions of Asian North and Far East and comprehensive feasibility studies were performed with the aim to determine the role and scale of SNPPs use in these regions. By the beginning of 1986 based on the results of this work the program of small nuclear power development had been prepared. This program was agreed upon with the industry and envisaged the detailed substantiation of SNPP construction at 33 particular sites of northern Russian regions.

The Chernobyl accident resulted in a sudden suspension of all plans for nuclear power development in the country, naturally including the SNPP construction program. It caused the designs of nuclear reactors and plants to be thoroughly analyzed and upgraded in order to meet the present–day domestic requirements and regulations, as well as IAEA recommendations in regard to safety improvement, nuclear weapons non–proliferation, etc.
During this period more than twenty reactor options and SNPPs have been studied in the field of small nuclear power, including conceptual designs for: water–cooled reactors AST–200, AST–30B, ATES–80 (150, 200), NIKA–150, RUTA–10 (20, 30, 55), lead–bismuth cooled reactors “Angstrem”, SVBR–75 and a modular fast neutron reactor with sodium coolant, BMN–170.

Preliminary engineering studies and conceptual designs of nuclear reactors have been prepared for unattended nuclear power plants of a very low capacity (1–2 MW (e)), e.g. “ELENA”, SAKhA-92, UNITERM, KROT.

In 1991–94 under the auspices of the Russian Federationn Nuclear Society the contest called “SNPPs-91” was held (Table II.)

TABLE II. RESULTS OF “SNPPS-91” CONTEST

Rank	Reactor unit power range (MW(t))						
	Below 10		10-50			Above 50	
	DHP*	NCGP**	DHP	NCGP stationary	NCGP floating	NCGP stationary	NCGP floating
1	“Elena”	-	RUTA	Angstrem	ABV-6	NCGP -80	KLT-40
2	-	Sakha-92 KROT	-	ABV-6	NIKA-120	-	NIKA-500
3	-	TES-M	-	ATU-M	-	ABV-13 VK-25	-

* DHP - District Heating Plant
** NCGP-Nuclear Co-Generation Plant.

3. CURRENT STATUS OF SNPP

By now the reactor designs of a new generation have been developed for SNPPs. They are much more reliable and safe as compared to the existing large NPP reactors in operation. That is because for small power reactor designs it appears possible to implement approaches which form the basis for the current concepts in nuclear and radiation safety, i.e. development of self–protection of reactor (reactors with inherent safety properties), use of passive self–actuating safety systems, defense-in-depth and leak-tight barriers of possible radioactive product release.

The following designs of reactors and SNPPs are currently readily available or could be available in the short term:

- KLT–40C, ship propulsion PWR-type reactor. The total service lifetime of such reactors being in operation on nuclear ice–breakers and on the light–carrier “Sevmorput” exceeds 150 reactor–years. In 1996 the design of nuclear floating power unit (NFPU) with two KLT-40C reactors for a nuclear co–generation plant (NCGP) was started. Nowadays the

NFPU basic design and licensing for the construction of the first–of–a–kind NFPU are in their final stage. The NFPU is planned to be constructed and located at the Northern Machine–Building Plant (Severodvinsk, Arkhangelsk region). The FPU installed capacity is 2 × 35 MW(e) and its heating capacity is 2 × 50 Gcal*h^{-1} [4], [5]

- ABV–6 (ABV–67–01) integral PWR reactor. In 1993 the basic design of floating NCGP "Volnolom–3" with two ABV–67–01 reactors was accomplished. Construction work was suspended for financial reasons. Currently site selection is under way with the aim to implement the project for the period up to 2010. The floating NCGP installed capacity is equal to 2 × 6 MW(e), its heating efficiency is 2 × 12 Gcal.*h^{-1} [6]
- ATU–2 water–graphite reactor. In 1996 the basic design of the second stage of the Bilibino NPP with three ATU–2 reactors was developed and reviewed by regulatory bodies. Four power units with EGP–6 reactors being in operation at the BiNPP since 1974–76, are the operating prototype of ATU-2 reactors. The BiNPP–2 installed capacity is equal to 3 × 40 MW(e) and 3 × 50 Gcal*h^{-1} of heat for district heating [7].

It is also worth noting two more SNPP designs which seem promising from the point of view of those conditions in which power engineering will be developed in the North and Far East and in other fuel–deficient regions of the Russian Federation:

- Floating NCGP "Crystal" with a small draught (~ 2,5 m) when it's being towed to the place of location. This plant design is based on the designs of ABV–6 reactor and floating NCGP "Volnolom–3" [6]
- DHP with RUTA reactors, these are low–temperature pool–type reactors with the atmospheric pressure in the reactor pool. The currently operating pool–type research reactors constructed in many places (more than 400 reactors) including the developing countries, can serve as prototypes of these reactors These reactors are in operation in the Russian Federation, Belorussia, Kazakhstan and other CIS countries [8].

At present SNPPs are being developed in line with a number common requirements:

- to use simple and reliable design solutions tested in practice
- to solve safety problems by using, to a significant extent, inherent safety features and passive safety systems
- nuclear weapon proliferation resistance, technological support to non–proliferation
- to provide a highly reliable protection against radioactive contamination of the environment in any operation conditions; environmental clean technology
- simplicity of control with the minimum number of operating staff
- maximized in-shop fabrication
- transportability to the operation site
- evacuation after shutdown and complete restoration of the occupied area.
- reliable operation in during daily variations of electrical load
- economic competitiveness.

The SNPP designs cover a very broad power range; there is a hundred times the difference between the lowest and the highest power of SNPP reactors. It is quite natural then that specific design requirements, safety requirements, operation and maintenance approaches for SNPPs are grouped in terms of smaller power ranges; below 10 MW, 10–50 MW, etc.

4. DEVELOPMENT PROSPECTS

The USSR disintegration and radical changes in political and social systems in the country resulted in changes to economic relations, including the industrial sphere. It also caused aspirations of regions for their economic independence and certain tendencies to significantly change economic specializations of some regions, including the northern ones. All these factors can lead and have already led to appreciable reduction in the demand of these regions for electricity and heat. For example, it can be clearly seen from the data on energy consumption during the last 10 years in the Chaun–Bilibino power grid. All these things make it necessary to revise the immediate plans worked out earlier as well as long–term prospects for constructing SNPPs. However, with all these facts in mind, power supply to the fuel–deficient regions still remains the principal goal in developing small nuclear power engineering in the country, especially in the area of decentralized power supply, and most of all for the northern regions.

Besides the North regions, there are certain prospects to use SNPPs, including nuclear heat–generating plants, for heat supply to towns, settlements and industrial facilities located in the areas of centralized power supply but still suffering continuous shortages of fuel supply for heating. For instance, some regions of Far East, Northern regions of Yakutia and the European part of the country, should be referred to this area.

In the future seawater desalination and demineralization of water with a high content of mineral impurities and salt (brackish water), can become one of the potentially important applications of SNPPs. Moreover, it might be possible to enter the market of developing water–deficient countries [9].

The USSR is a pioneer in the field of using nuclear energy for fresh water production by means of seawater desalination. In 1973 on the east coast of the Caspian Sea in the town of Shevchenko (now Aktau, Kazakhstan) the first, and, as it turned out, the only in the world industrial water desalinating complex based on the BN–350 sodium–cooled reactor was put into operation. The successful experience of its operation, many desalinating plant designs with various reactor types, and comprehensive studies of this issue within the framework of IAEA research programs give grounds for considering nuclear reactors as promising power sources for desalination plants. The studies have revealed a stable interest in the use of small reactors, which are capable of producing fresh water on vast territories with decentralized local power supply, typical of many water-deficient regions of poorly developed countries. The latest examples of the activity of Russian specialists in this direction are quoted in the papers, which are presented at this seminar [10, 11].

The analysis for the prospects of using SNPPs to cover deficiencies in power capacities in remote regions, isolated from the centralized power grids, has confirmed that SNPPs should be considered as efficient electric and heat sources when choosing power supply for isolated fuel–deficient regions of North and Far East. SNPPs should be also considered as a potential product for the international market. At the same time it is clear that in the XXIst Century the unique fossil and renewable natural resources of northern regions of the country will be in intensive demand. Hence, the principal targets and prerequisites for developing small nuclear power in our country remain the same.

CONCLUSIONS

In conclusion it could be stated that the use of SNPPs in the fuel–deficient regions gives the possibility to limit and in some regions even reduce the consumption of deficient, expensive organic fuel which has to be transported from afar. The operation of the Bilibino NCGP is a good example of that. At the same time providing a much higher quality of energy supply to industrial facilities and residential areas and thus, improving social and cultural living standards of population seems possible. So, in a number of Far East and North regions the use of NPPs in combination with other types of power sources can result in the required progress in all the spheres of life and human activity and in the solution of economic, environmental and social problems.

That is why the main and paramount objective of developing and using SNPPs in the Russian Federation still consists of reliable power and heat supply to public utilities and industrial consumers located in remote, isolated, hard–to–reach, fuel–deficient regions of the country. These regions primarily refer to the area of decentralized power supply, which covers about 10 million km^2, i.e. more than 60 % of Russian territory. First of all these are Arctic areas of the North.

The current economic situation in the country undeniably hampers the solution of problems related to small nuclear power development. At the same time we cannot do without the solution to these questions because in a number of regions of the North and Far East only the use of NPPs, in combination with other power sources, will be able to provide the required progress in industrial activity of people, improvement of living standards, social and cultural development.

So the small nuclear power development, i.e. SNPPs construction and operation will be an effective tool and important condition for providing economic and social progression of the Russian regions under consideration. In view of that, today it is clear that the goals in the field of small nuclear power still remain sufficiently urgent and pre-requisites for their efficient implementation by means of SNPPs are important.

The general policy in the field of small nuclear power engineering for the future is formulated in the following documents: "Program of nuclear power development in Russian Federation for 1998–2005 and for the period up to 2010" and "Nuclear power development stategy in the Russian Federation in the first half of XXI century". In these documents it is stated that SNPPs design and construction, including the use of ship–propulsion technologies, is an important trend in the nuclear power development. The commissioning of the first floating NCGP with KLT-40c reactors is planned for the period of 2006–2010 as a specific task. In addition, before 2010 an SNPP of enhanced safety is planned to be designed for the peripheral regions of the country, together with the infrastructure for its operation and maintenance. A number of installations of small nuclear power are also planned to be constructed before 2030, including stationary, floating power and desalinating plants.

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PANEL DISCUSSION

(Summarized by D. Majumdar)

Several issues were discussed; notable of these were local participation, infrastructure and non-proliferation.

It was mentioned that nuclear energy is not only important for energy production, but also the spin off effects of nuclear industry are important for the general industrialization of a developing country. Local participation is significant from this point of view. The concept of floating nuclear power plants was used to highlight this subject because they have minimal requirements for local participation. It was said that local participation could be negotiated and made optimal. The experience of several developing countries has been that the first plant has been bought on a turnkey basis with minimum national participation. Even in such cases local contribution is around 15 – 20 % consisting of local labor, civil works contracts, etc. The advantage of having a floating reactor, however, is that it accelerates the introduction of nuclear power in the country's grid rather than continuing to build fossil plants. In most developing countries the generation growth is happening mainly through fossil plants. For example, in India some 8000 Mw growth is taking place every year through coal and similarly in China, about 12,000 to 15,000 Mw growth every year through coal. So turnkey nuclear plants could play a significant role.

Despite the very little scope for local participation, the concept of a floating nuclear power plant has obvious advantages such as higher quality of in-shop assembling, shorter construction time, flexible siting, etc. These factors have also to be weighed and assessed in the decision making process. Besides, on-shore facilities and civil works are needed even for a floating plant. Local participation in this area can be rather broad, depending on the available industrial capability of a user country. But safety and quality must be given the highest priority.

There was a question regarding why proliferation-resistant reactor designs are so important now. The concern came from the possibility that a large number of nuclear plants could be built in many countries around the world. Ensuring and verifying non-diversion of materials to any weapons related programme become a serious concern. So if we could make the designs proliferation-resistant, the concern is minimized, and hence there should be this additional push to increase the proliferation resistance of the technologies. It should be an inherent part in the design, in the capability of the system; and this can be done in several different ways - in the fuel cycle, restricting access to the fuel, operating longer cycles, burning as much plutonium as one produces, etc. There are technologies available for these purposes that we haven't utilized in the past. But with the deployment of larger number of reactors this issue has to be addressed as best as we can, not only through safeguards, but also through the designs of the systems.

Another perspective on this subject was also discussed: that development of new proliferation-resistant designs should not stand in the way of bringing the stout technology for improving the climate and for the economic growth of the developing world. In other words, the development of new proliferation resistant technologies should not be the precondition for moving forward as rapidly as possible where it can meet human needs.

ECONOMICS AND FINANCING

(Session 3)

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ECONOMIC AND FINANCING ISSUES OF NUCLEAR POWER — INDIAN EXPERIENCE

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Abstract

This paper deals with the Indian experience related to economic and financing issues in implementing a nuclear power programme successfully. It highlights the ingredients that have resulted in the success of the programme. It suggests that the model adopted by India can also be adopted by other developing countries with suitable modifications and proposes certain measures to help the implementation of a nuclear power programme in developing countries.

1. INTRODUCTION

During the last 5 decades India has seen its installed capacity of power growing at a compounded rate of 7% per year which has been about 2% higher than the economic growth rate of the country during the same period. Nuclear power capacity on the other hand, which emerged with the commissioning of two units of Boiling Water Reactors (BWR) at Tarapur has been growing at a pace slower than the power sector as a whole.

Considering the low per capita consumption of electricity in India, the power sector, which has a capacity of about 100,000 MWe at present, is expected to grow at a high rate for the next twenty to thirty years. The present share of different sources in the electricity generation in the country, as of now is given below:

TABLE I. SHARE OF DIFFERENT SOURCES

Sources	Present share
Thermal	72
Hydro	24
Nuclear	3
Other	1
Total	100

The State Electricity Boards (SEBs) which deal with the generation and distribution of electricity in each state and are controlled by the respective state government, are at the center stage of the power sector in India. They together have a share of about 60% of generation and almost 100% of distribution of electricity in the country. Due to the heavy subsidy in tariff provided by the SEBs for supply of electricity to the agricultural sector and also due to the high level of losses in transmission and distribution, most of the SEBs are not in a position to recover the full cost of the electricity they supply which has resulted in the deterioration of the financial position of the SEBs.

1.1. Deregulation

The economic reforms initiated in the country encompasses deregulation of the power sector, with a view to channelise private investments in the sector. However, the actual private investment that has materialised so far is far short of expectations. Presently the generating capacity in the private sector is only 9% of the total capacity. The breakup of category wise ownership of present generating capacity in the country is given in Table II.

TABLE II. OWNERSHIP OF CAPACITY OF POWER

Category	Percentage of capacity
State Electricity Boards owned by the State Governments	61
Central Generating Units owned by the Central Government	30
Private Sector	9

Source: Annual Report 1999-2000 of the Ministry of Power, India.

2. GROWTH OF NUCLEAR POWER

India was an early entrant in the field of nuclear power by setting up two units of BWR at Tarapur in 1969 based on enriched uranium as the fuel. The Tarapur units were set up as a turnkey contract to demonstrate the viable operation of nuclear power plants in the country. Thereafter, through a collaborative venture with the Atomic Energy of Canada Ltd. (AECL), two units of pressurised Heavy Water Reactors (PHWR) (2 × 200 MWe) were installed at Rawatbhatta in Rajasthan, where local content was employed in the form of equipment, materials and field management supervision. Subsequent to the installation of the two PHWR units at Rawatbhatta, a series of similar units were built at different parts of the country entirely with indigenous efforts.

The corner stone of Indian nuclear power programme was the indigenisation of the technology, which has its benefits as well as costs. Indigenisation of the technology has helped in developing the industrial infrastructure in the country and while supporting nuclear power to achieve a stage of maturity, there have been costs in terms of longer gestation period for the initial projects and consequent cost overruns. In addition, the units which were commissioned from time to time, encountered various operational problems which were to be resolved entirely by local efforts, taking more time, which had an adverse effect on the performance levels of the units.

The gradual increase in the local content of the nuclear power plant units built in India is shown in Table III.

TABLE III LOCAL CONTENT IN SUCCESSIVE REACTOR UNITS

RAPS-1	RAPS-2	MAPS	NAPS	KAPS	Kaiga
54%	75%	80%	92%	90%	95%

Source : Cost Data of NPCIL

2.1. Research and Development (R&D) efforts

It was recognised even at the beginning of the nuclear power programme, that a well developed R&D infrastructure was essential for the smooth absorption of the imported technologies and also resolution of the operational problems as well as upgrading of the technologies to more advanced versions. Apart from the dedicated R&D facilities under the Department of Atomic Energy, R&D support for nuclear power is also drawn from various educational and research institutes in the country.

2.2. Regulation of Nuclear Power

Atomic Energy Regulatory Board (AERB), a body independent of the agency executing and operating nuclear power stations, supervises the regulation of construction, commissioning and operation of nuclear power stations. The regulatory processes have been evolved over a period of time and are getting streamlined, which would facilitate safe as well as smooth operation of nuclear power stations.

2.3. Public Awareness Programme

Having realised that public support is an essential ingredient for the development and growth of nuclear power in the country, a public awareness programme was initiated in an organised manner. The objective of the programme is dissemination of scientifically correct and authentic information in order to remove mis-apprehensions of the public at large about the nuclear power programme especially about the safety, economy, environmental impact, waste management etc.

The programme has been quite effective in mobilising public support for the nuclear power programme. The public at large in India has been supportive of nuclear power as it is perceived as a source of energy with a large potential and competitive in costs with other sources of energy. However, there is a section of the population critical of nuclear power mainly due to the safety issues involved.

3. ECONOMIC ISSUES

Economics of nuclear power varies form country to country depending on the availability and costs of other sources of energy. The coal based thermal power stations generate more than 65% of the total electricity in India. Therefore, economics of nuclear power is generally compared with that of coal based thermal power. Even though coal based thermal power stations are located all over the country, the coal deposits are limited to certain parts of the country, particularly in the eastern region. Therefore, for the stations located in the northern, western and southern regions of the country, coal may have to be transported over long distances from the pit -head. Cost of transportation is a significant part of the fuel cost for coal based thermal power depending on the distances of transportation. Therefore, the economics of coal based thermal power has a significant bearing on the locations of the stations.

In the 60s and the 70s, nuclear power with somewhat higher capital costs and substantially lower operating costs than thermal power was seen as a cheap source of power. However, the scenario gradually changed with the increase in the capital costs of the successive units to incorporate more and more safety features to upgrade the designs to the international safety standards. Combined with this, the increase in the operating costs eroded the economic advantage of nuclear power.

In India, based on the present levels of generation costs, nuclear power is competitive with coal based thermal power, at locations of about 1000 km away from the pit head. Clean coal technologies are needed to mitigate the adverse environmental effects of fossil fuels. These technologies are still to be developed, which when developed and employed could increase the capital costs per MW of coal based units by more than 25% and thereby enhancing the economic advantage of nuclear power. In addition, there is substantial potential to improve the economics of nuclear power by controlling the capital costs and also by improving the operational performance.

3.1. Capital Cost of Projects

Substantial increases in capital costs have been observed in the successive reactor units mainly due to inflation. However, even at constant currency value, there have been significant increases in the costs particularly for the reactor units completed in the nineties and later, due to the contribution of financing costs as well as due to incorporation of improved safety features. The strategy for reduction in capital costs consists of:

- standardisation of the unit and serial construction
- scaling up the units size to an optimum capacity.

It is expected that with this strategy, the capital costs per MW could be reduced in real terms by about 35% from the current levels.

3.2. Tariff and Return on Investments

Electricity generated by the nuclear power stations are supplied to the SEBs for which the bulk power tariff is fixed. The tariff is based on **'cost plus'** principle and the norms for formulation of the tariff are notified by the government of India. As per the norms prevailing now, a return on equity of 16% after meeting all the costs including depreciation and interest on debt capital can be earned at the normative performance level of operation of the stations. If the station is able to operate at a higher level than the normative performance level, the return will be more than 16%. The return is based on the original book value of the investments and therefore, due to inflation, the value of returns in real terms gets diminished over the years.

4. FINANCING

Investments in nuclear power in the initial phase of the programme has been entirely funded by the government through its national budgets. With a view to mobilise funds from sources other than the government and also to implement capacity addition in nuclear power in an expeditious manner, Nuclear Power Corporation of India Ltd. (NPCIL), a company fully owned by the government of India was incorporated in 1987. Since its formation, NPCIL has been able to mobilise debt capital from the Indian capital market for part financing the capacity addition in the nuclear power sector. The investments made in the nuclear power sector so far is about Rs.104 Billion of which Rs.46 Billion is borrowed from the capital market. The investments at today's money value is about Rs.250 Billion of which about 35% is borrowed funds.

4.1. Indian Capital Market

The Indian capital market has been growing at a high rate during the past ten years and is expected to continue its growth in the future as well. A part of the domestic saving, which at present is about 23% of the GDP, is channelised into the capital market and deployed in capital formation. The capital market in India offers a great opportunity for funding nuclear power growth in the country.

4.2. Debt Equity Ratio

Debt capital for funding nuclear power growth has been mobilised only after the formation of NPCIL. Considering the uncertainty associated with the gestation period of the projects and also on the performance levels of operations of the nuclear power stations, the company had adopted a conservative debt equity ratio of 1:1. Even with this conservative debt equity ratio, in some of the projects where there were substantial increases in gestation periods, the financing costs of projects propelled to as high as 40% of the project cost as compared to about 20% that was originally estimated. A larger debt equity ratio has an impact on the capital cost due to higher interest during construction and thereby the unit energy price of electricity. At the same time, a larger debt component in the capital has a leveraging effect on the financial performance of the station thereby magnifying the profit when the performance level of the station exceeds the normative level. More over, the debt component in the capital is redeemed gradually which will have an effect of gradual reduction of the tariff for power. Therefore, higher debt component in the capital could be preferred when the project is executed with optimum gestation period and the station is expected to perform at high operating levels on a continuous basis.

Nuclear power technology has attained a level of maturity in India and the stations have been operating at high operating levels on a sustainable basis. There are indications that the gestation period of the future projects would be reduced to an optimum. It is therefore possible and preferable to use larger debt capital to the extent of about 70% in the future capacity addition. The present growth trend of the Indian Capital market suggests that it could meet the growing demand for debt capital for the future growth of nuclear power in India.

4.3. Improvement in Operating Performance

Until the mid- nineties, the annual average plant load factor of operating nuclear power stations was at about 60%. With the continuous efforts in improving the performance levels by efficient outage management and also by preventive maintenance, the performance levels of the stations have gradually improved year after year. With the increase in the operational performance of the stations, the internal surplus generated by the company has also increased as can be seen in Table IV.

TABLE IV. INTERNAL SURPLUS GENERATED BY NPCIL (RS BILLION)

1990-91	**91-92**	**92-93**	**93-94**	**94-95**	**95-96**	**96-97**	**97-98**	**98-99**	**99-2000**	**2000-01**
0.96	0.55	1.13	(-) 0.86	(-) 0.48	2.43	4.52	4.61	5.70	6.50	14.23

4.4. Credit Rating of Bonds

The bonds issued by NPCIL were first credit rated at A- in 1995, which indicates adequate safety. Since then, the performance of NPCIL in terms of electricity generation and profits earned has improved year after year which in turn improved the credit worthiness of the company. This improvement in the credit worthiness of the company has been reflected in terms of improvement in the credit rating of its bonds which is shown in Table V.

TABLE V. IMPROVEMENT IN THE CREDIT RATING OF NPCIL'S BONDS

1995-96	**1996-97**	**1997-98**	**1998-99**	**1999-2000**	**2000-01**
A-	A-	A	A+	AA	AAA
Adequate Safety	Adequate safety	Adequate safety	Adequate safety	High Safety	Highest Safety

4.5. Joint Venture for Capacity Addition

Nuclear power technology has attained a level of maturity in India. With the good performance of nuclear power stations and consequent generation of enhanced internal surplus, the confidence level of Indian corporates in the nuclear power business has improved. Some of the corporates engaged in the power sector, have shown interest in participating in a joint venture with NPCIL for building new nuclear power projects. NPCIL is examining the possibilities of forming a joint venture with the participation by Indian corporates for building new nuclear power projects.

4.6. Indian Model

The Indian model on nuclear power financing has the following features:-

- Selection of technology & unit size to optimally utilise indigenous capabilities,
- Initial growth entirely funded by the Government,
- Indigenisation of the technology with maximum participation from local industries with high level of R&D support and upgrading of local industrial infrastructure,
- Focusing attention on good performance of nuclear power stations by better outage management and preventive maintenance to improve economic viability of nuclear power,
- Gradual accessing of local capital market for the debt finance with improved credit worthiness for nuclear power,
- Increasing the debt content of the capital as the performance of the plants improved,
- Continuous programme with standardisation and serial construction as the strategy to achieve significant contribution from nuclear power at competitive costs,
- Action to form joint venture companies with participation from the private and public sector companies to speed up capacity addition,

5. NUCLEAR POWER IN DEVELOPING COUNTRIES

Any developing country embarking on a nuclear power programme could also adopt the model followed by India with suitable modifications. A high rate of demand growth for electricity is expected in all the developing countries and a part of such demand growth could be met by nuclear power. Considering the size of the electricity grid and the level of industrial infrastructure existing in the developing countries, Small and Medium Power Reactors (SMPRs) are the appropriate range of units for the developing countries. However, the main obstacles in realising nuclear power growth in a developing country are scarcity of capital and non-availability of technology.

Nuclear power technology is well established in some of the developed countries. Further growth of nuclear power in the developed countries is not expected to be significant in the immediate future. The nuclear industrial infrastructure established in the developed countries could gainfully support nuclear power growth in the developing countries by way of supplying technology and equipment. However, the main challenge faced by the developing countries in building nuclear power is the scarcity of capital.

A part of the costs of a nuclear power project of the order of about 30% could be funded through the suppliers' credits which are re-financed at concessional rates of interest by the export promotion agencies in each country to promote exports from that country. For the balance costs of the project, however, other sources of funding have to be found. Unlike conventional sources of power viz. thermal and hydro, international institutional financing is not available for nuclear power projects. Creation of an international fund for financing the initial growth of nuclear power in the developing countries could mitigate this problem. This fund could be managed by the International Atomic Energy Agency (IAEA), either on its own or in association with an international financial institution.

The objective of the fund is to finance the initial growth of nuclear power in the developing countries. The fund could provide project finance up to certain percentage of the project costs, over and above the financing available through the suppliers' credits. Before extending support from the fund, the viability of each project proposal may be established to ensure that there exist a system of efficient distribution of electricity from the station to the consumers, a system of fixation of tariff reflecting all costs and a system of recovery of electricity charges from the consumers. Based on the viability of each nuclear power project, a decision could be taken to extend a loan from the fund to the power project. In order to guarantee repayment of the loan, it could be channeled through the respective government of the developing country which could absorb the exchange rate variations and extend the loan to the project at a fixed rate of interest. The track record of repayment of the loan for one project could be a criterion to be considered for extending a second loan for another project in the same country.

The growth of nuclear power in the developing countries would also benefit the nuclear industrial infrastructure already established in some of the developed countries in terms of getting orders for equipment and technological services. Further, growth of nuclear power in the developing countries would help in controlling emissions of the greenhouse gases, the benefit of which will accrue to all. In view of the above and also to reduce the average cost of the fund, some of the developed countries may consider long term contributions to this fund at concessional interest rates. In addition, resources could also be mobilised from the international capital market. Once the fund has attained a certain critical size, it could be self sustaining.

With regard to the movement of technology and equipment from the developed countries to the developing countries for nuclear power growth, it is proposed that a frame work of procedures may be established and co-ordinated by the IAEA. The agency can recommend the technology and plant which are suitable for each country, based on its population, electricity demand and the local industrial infrastructure. In order to improve the economics of nuclear power in the developing countries, maximum local content, without compromising on quality, should be used for the nuclear power projects.

CONCLUSION

Financing for capacity addition for nuclear power in India, which until a few years ago was entirely through its national budgets, gradually changed with the improved performance of the nuclear power stations and the growth of the Indian capital market. From a position of 100% government support, the nuclear power stations today derive up to 50% financing from the capital market. The success of this financing model was due to successful indigenisation of the technology, excellent performance of the operating nuclear power stations leading to generation of substantial internal surplus thereby creating confidence in the business circles about the financial viability of nuclear power. This model with suitable modifications can be adopted in a developing country embarking on a nuclear power programme. Creation of an international fund for nuclear power development will facilitate growth of nuclear power in the developing countries.

CAN SMALL AND MEDIUM SIZED REACTORS BE COMPETITIVE?

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Abstract

Cost is an important attribute for the future development of small and medium sized reactors (SMRs). This paper addresses the target costs necessary for SMRs to be competitive in the markets of the 21st century and ways and means that designers are considering to meet these targets. It briefly reviews the impacts on nuclear energy competitiveness of economic globalisation, market deregulation, privatisation of the power sector and policy measures aiming at sustainable development and highlights new challenges and opportunities that are created for nuclear power plants as a result of the changing policy making framework in the electricity sector. Recognising the importance of capital cost (60% or more of nuclear electricity generation cost) the paper elaborates on technical (e.g. simplified design and factory building) and managerial (e.g. series order) measures aiming at capital cost reduction. The impacts of deregulation of the electricity market and privatisation of the sector on the competitiveness of SMRs are addressed. The paper stresses the importance of internalisation of social and environmental costs and sustainable development objectives for the competitiveness of nuclear energy and in particular of SMRs that are adapted to decentralised electricity supply and co-generation. Finally, it presents some findings and conclusions, drawn from the NEA analyses, on the prospects for enhancing the economics of small and medium reactors and their potential contribution to sustainable energy mixes.

1. INTRODUCTION

Decisions on technologies and energy mixes for electricity generation have to take into account a variety of non-economic issues, including technical factors, social acceptability, and health and environmental impacts. Nonetheless, economic competitiveness of different options remains a dominant factor in the decision-making process and, if an option is not economically viable, none of the other factors is likely to lead to its implementation. Therefore, the question of whether small and medium sized reactors (SMRs) can be competitive is key to their development and eventual deployment.

The ongoing changes in the policy-making framework of the electricity sector are likely to create new challenges and opportunities for nuclear energy in general and for SMRs in particular. In order to assess the current and future economics of SMRs, it is relevant to review briefly the impacts on nuclear energy competitiveness of electricity market deregulation, privatisation of the power sector, and policy measures aiming towards sustainable development.

Recent economic studies on electricity generation costs, including the 1998 OECD report on projected costs of generating electricity, give some insights on the costs that would make small and medium sized reactors competitive on present and future markets. In turn, the results from these studies provide designers and manufacturers of SMRs with targets to be met in order to develop technologies and products that may be successfully deployed at the industrial and commercial scale.

Capital cost represents some 60% or more of nuclear electricity generation cost. The importance of this component justifies investigating technical and managerial measures aiming at reducing capital costs of small and medium sized reactors. In this context, it is worth noting that the specific features of SMRs allow to take advantage of simplified design, factory building of plant sub-systems and series ordering, as means to compensate for their lack of economy of scale in comparison with larger nuclear units or classic fossil-fuelled power plants.

International economic analyses show that market competition will be difficult for nuclear energy in the coming decades and small and medium sized reactors will be no exception. However, small and medium sized units offer some advantages that may be determinant for their successful deployment provided sustained research and development efforts are pursued in order to lower their costs, in particular the investment part, while maintaining a high level of safety and excellent technical performance.

2. IMPACTS OF THE NEW POLICY-MAKING LANDSCAPE ON SMR COMPETITIVENESS

Trends to deregulation of electricity markets and privatisation of the sector may have significant impacts on the relative competitiveness of alternative energy sources and generation technologies [1]. Deregulated electricity markets are expected to enhance economic efficiency and promote the development of the cheapest technologies. The future deployment of small and medium sized reactors in competitive markets will be possible when and where their economic advantages as compared with alternatives (i.e. fossil fuels, large nuclear units and eventually, in some locations, renewable energy sources), are demonstrated.

Privatisation of the electricity sector creates a challenge for capital intensive power plants since private investors are looking for projects that mature reasonably rapidly. In general, nuclear power plants have longer planning and construction times and higher investment costs than other generation technologies. Therefore, nuclear power projects face particular challenges in a privatised electricity sector. However, the challenge of keeping investment costs acceptable for private investors is less difficult to address for SMRs than for large-size nuclear units because their total investment costs are lower and their construction periods are expected to be shorter.

Since long-term demand forecasts for individual utility sales are more uncertain in a deregulated market than in monopoly situations, projects with large-scale investments and long implementation times could lead to unacceptable financial risks because of the possibility that sales might be lower than expected. Therefore, private investors may wish to invest in technologies that can be implemented in smaller increments of capacity, within shorter periods of time and that have smaller total investment costs. Small modular reactors may become rather attractive in this context, especially if new advanced designs continue placing emphasis on concept simplification and streamlining, leading to easier and quicker construction as well as lower overnight capital costs.

Also, competitive electricity markets are anticipated to need power supply flexibility to accommodate customer requirements. Therefore, power companies may wish to spread their investments among different fuel sources and plant types. The introduction of nuclear power may become a key element of supply diversification in this context. Furthermore, nuclear power plants contribute to long-term stability of generation costs since nuclear fuel costs are lower and less volatile than fossil fuel prices. Small and medium sized reactors offer opportunities in this regard at limited risks owing to the possibility to diversify a system without adding a large increment of capacity.

The other main evolution in the decision-making landscape which is creating new challenges and opportunities for nuclear energy is the integration of sustainable development goals in national policies. According to the analyses carried out by the IAEA [2] and the NEA [3], the characteristics of nuclear energy are generally compatible with sustainable development objectives. In terms of achieving sustainable development goals, SMRs may offer some advantages as compared with large nuclear power plants, for example as a source of decentralised generation able to provide energy services to developing countries and rural areas. However, the construction and operation of a large number of SMRs on multiple sites may raise some social and environmental concerns from a sustainable development view point.

In economic terms, the increasing awareness of environmental issues and the recognition of broad macroeconomic and social effects arising from technology choices are leading to new approaches and additional criteria in the comparative assessment of different generation options. Cost comparisons of generation technologies can be taken beyond the traditional approach of calculating the direct economic costs to the utility through internalising other costs to society, i.e. externalities, insofar as feasible.

Internalising externalities might enhance the competitiveness of nuclear power versus coal- and gas-fired power plants. Owing to the early recognition of the need to adequately protect the public and the environment from ionising radiation, the classic levelised cost assessment already takes into account most of the elements related to health and environmental impacts of nuclear power generation, from mining through electricity generation to decommissioning of the facilities, waste management and disposal. Also, the costs related to the application of safety standards and regulations are embedded in the investment, operation and maintenance costs of nuclear power plants.

Small and medium sized reactor projects that would be considered in this context would benefit from a comprehensive comparative assessment of full cost to society of alternative options. Since the externalities arising from fossil fuel electricity generation, for example the potential costs of greenhouse gas emissions, are not taken fully into account at present, their recognition and internalisation would increase the costs of fossil fuel based generation relative to nuclear power plants, including SMRs.

3. TARGET COSTS FOR SMRS

For uses other than electricity, such as cogeneration, heat supply, water desalination and eventually hydrogen production, it is difficult to assess the economics of SMRs versus alternatives on a generic basis. Costs for this type of projects are essentially site specific, depend on local conditions and are sensitive to the demand for heat, water, etc. Therefore, cost comparisons between alternatives should be made on a case by case basis. The prospects for SMRs to be the cheapest option will vary from country to country and from application to application. A number of case studies carried under the auspices of the IAEA show that SMRs may be competitive for non-electrical applications under certain conditions.

For electricity generation, the SMRs that are currently being developed could be commercially available for commissioning by 2005-2010. Their main competitors will be state-of-the-art gas-fired and coal-fired power plants, as well as large nuclear units of the evolutionary type. The economic studies carried out by international organisations, such as the NEA and the IAEA, on generic costs of generating electricity provide some insights on the targets that SMRs should achieve to be competitive.

The latest OECD study on projected costs of generating electricity [4], published in 1998, is based upon data provided by fourteen OECD countries and five non-member countries invited by the IAEA. Projected costs of generating electricity were estimated in the study using cost elements corresponding to 72 power plant projects. Total electricity generation costs were calculated using the levelised lifetime cost methodology and common generic assumptions for some key parameters. In particular, for all types of power plants considered the study assumed a 40 year economic lifetime, 75% availability factor and two reference discount rates, 5% and 10%.

The results from the study indicate that projected generation costs for coal-fired power plants range between 25 and 45 mill/kWh[1] at 5% discount rate and between 35 and 60 mill/kWh at 10% discount rate. Those costs correspond to coal prices, provided by participating countries, ranging from 1 US$/GJ to 2.8 US$/GJ in 2005 - year of commissioning of the plant - and increasing at an average escalation rate of 0.3% per annum.

For gas-fired power plants, the range of projected generation costs is 25 to 50 mill/kWh at 5% discount rate and remains similar at 10% discount rate since the low capital costs of gas-fired power plants make gas-generated electricity costs rather insensitive to discount rate. The gas prices assumed vary between 1.6 US$/GJ and 5.4 US$/GJ in 2005 with a 0.8% per annum average escalation rate. In this connection, it should be stressed that since the study was carried out the international oil and gas market conditions evolved significantly, leading to steep increases of gas prices in many countries.

For nuclear power plants, mainly large size units of the evolutionary reactor type, the ranges of projected generation costs are 25 to 40 mill/kWh and 40 to 60 mill/kWh respectively at 5% and 10% discount rate. The overnight base construction costs of nuclear power plants considered in the study, provided by participating countries, ranged between just over 1 000 US$/kWe and more than 2 000 US$/kWe. Adding the interest during construction over a period of 5 to 9 years and provisions for decommissioning, the total capital costs of the nuclear units considered in the study ranged between 1 400 and 2 500 US$/kWe at 5% discount rate and between 1 700 and nearly 3 000 US$/kWe at 10% discount rate.

The study showed that, at a 5% discount rate, the least expensive (by a margin of at least 10%) is gas in three countries, coal in three countries, and nuclear in five countries. In seven countries, no option is has a competitive margin of at least 10%. At a 10% discount rate, the least expensive option (by a margin of at least 10%) is gas in nine countries and coal in one country. Nuclear power is not found to be the least expensive option in any country. In eight countries, there is less than a 10% difference between the two cheapest options.

From the study results, it appears that no single technology has a clear economic advantage in all countries. Specific circumstances within each country will determine the most economic choice. A key factor regarding the competitiveness of nuclear versus gas-fired power plants, that are their main competitor today, is gas price escalation assumptions since a doubling of the gas price increases by some 70% the cost of gas-fuelled electricity. Recent trends, highlighting the volatility of oil and gas markets, undoubtedly provide incentives to revisit the nuclear option.

The ranges of generation costs for each technology/energy source are quite broad, underscoring the observation that competitiveness should be assessed on a case by case basis

[1] 1 mill = 10^{-3} US$.

at the country and utility level, based upon specific technical and economic conditions applicable in each case. This is especially true for projects being considered in developing countries where the economic conditions vary significantly from the average in OECD. Nevertheless, the average generation costs given above provide an indication of target costs in order for small and medium sized reactors to be competitive, i.e. less than 1 000 US$/kWe overnight capital cost and less than 25 mill/kWh generation cost.

4. OPPORTUNITES FOR SMRS

Economies of scale have, for many years, led reactor designers and utilities to move toward larger unit sizes for nuclear power plants, as well as for other generation technologies, and this has in general reduced the costs of electricity production. However, the changing market conditions are prompting many power producers to take a fresh look at the optimum unit size [5].

Uncertain demand growth, greater competition in the electricity market, and long lead times for commissioning large units define the operating environment today for many utilities. For them, it may be too risky to commit the capital investment needed to build a large nuclear unit that must be started many years in advance of the anticipated need for the capacity. Indeed, if that need fails to develop, or develops several years later than expected, it could leave the utility with expensive excess capacity on its hands and inadequate revenue to finance its investments.

Today's market climate requires a good match between capacity and demand, because a major mismatch in either direction carries substantial cost penalties. Building capacity in smaller increments may be one way to optimise the match; hence, the growing interest among utilities in the concept of smaller generation units, sometimes including modular factory fabrication and series production aspects.

A number of research or design team, in many countries, are investigating small and medium sized reactors of advanced concepts, evolutionary and innovative, aiming at the development of power plants economically competitive and adapted to the markets of the new millennium. The main advantages of SMRs that are expected to enhance their competitiveness as compared with large size nuclear units are:

- small power outputs that offer a better fit to medium sized power grids;
- good adaptation to low load growth situations;
- size allowing countries with no nuclear power experience to launch a nuclear energy programme earlier than with large plants;
- contribution to system reliability through improvement in loss of capacity probability (less capacity is lost when a smaller plant shuts down);
- possibility to distribute economic risk over several small projects; and
- lower absolute capital cost leading to smaller financial burden for each plant.

The opportunities for reducing capital costs of SMRs deserve particular attention since the experience acquired so far by the nuclear industry generally supports the view that the larger the size of the plant, the lower the specific capital cost per kWe installed. The target capital costs suggested by recent economic analyses, i.e. significantly less than 1 000 US$/kWe, highlights that teams developing SMR concepts and designs should place emphasis on reducing capital costs.

SMRs can benefit from several of the ways and means that have been identified to reduce capital costs of nuclear power plants [6]. Improved construction methods, reduced construction schedule, standardisation and construction in series are applicable to small units as well as to large units. In fact, economies of series production can provide more benefit to SMRs than to large size units and may compensate for lack of economies of scale.

Design simplification and reliance on natural circulation, as well as passive safety devices, can lead to significant capital cost reductions and are already considered in a number of advanced SMR concepts. The use of modular systems and of equipment pre-fabricated in factories and workshops rather than on site provides opportunities for reducing construction schedule and uncertainties regarding timely project completion.

In countries embarking in a new nuclear energy programme based upon SMRs, in particular in developing countries, enhancing local participation at various stages of design and construction of nuclear power plants, pursued essentially for capacity building purposes, may reduce capital costs as well as operation and maintenance costs. Provided adequate education and training is ensured up-front, reliance on local manpower can contribute to reducing costs and construction time.

Achieving better economic performance of SMRs can be facilitated also by procurement policies taking advantage of equipment and materials available from other industrial sectors. The required “nuclear quality” is not unique to nuclear power facilities and a number of equipment items produced for other industrial sectors could be used in nuclear power plants. Obviously, the series effect applicable to equipment produced for several industrial sectors generates significant cost reductions as compared with nuclear-specific fabrication.

CONCLUDING REMARKS

The renewed interest in small and medium sized reactors is demonstrated by the large number of countries considering such reactor types and by the dynamism of research and design teams working on the development of advanced SMRs. The policy-making context of the new millennium offers opportunities to a broader development of nuclear energy in particular in countries that have not yet implemented a nuclear power programme. SMRs could benefit from this revival of the nuclear option if their adaptation to the market and their economic competitiveness are demonstrated.

In order to have a chance on competitive markets, SMRs should have reasonably low construction costs, short construction schedule, assured licensability and predictable operating and maintenance costs. They should be designed for high operating availability and long plant lifetime. Meeting those objectives is necessary in order to attract potential investors and protect their interests. Advanced designs being developed in many countries do aim at satisfying those criteria but, in most cases, their performance needs to be demonstrated at the prototype level before considering their commercial deployment.

Economic studies and past experience show that technological breakthroughs occur continuously and that economic competitiveness varies according to the regions and periods considered. Today, small and medium sized reactors are seldom the cheapest option, although in some locations and for specific uses, including non-electrical applications, they may be the optimal choice. Tomorrow, provided sustained efforts are pursued by designers to enhance the performance of advanced SMRs, they may compete successfully in many more circumstances.

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THE ROLE OF NUCLEAR ENERGY IN SOUTHEAST ASIA

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Abstract

Southeast Asia needs financial and infrastructure bases for continuous development in the 21st century. The role of nuclear energy in that region in combination with other generating facilities is evaluated on the basis of total costs, emissions, wastes, and energy security. Capacity expansion plans are composed by a technique of dynamic expansion planning, which can realize minimum costs from the year 2000 through 2020 for Thai and Vietnamese cases. Although in many scenarios gas-fired power plants are more economically feasible than coal-fired and nuclear power plants, nuclear energy can play an important role for the best energy mix. Also, improvements of nuclear power plant's performance and its suitable unit size that make nuclear energy competitive enough are diagnosed with electric power systems. This study concludes that nuclear energy is possibly compatible with the energy, economic and environmental conditions that may prevail in this region.

1. INTRODUCTION

Asia, where about 60% of the world's population lives and about a quarter of the world total energy is consumed, is expected to grow more rapidly than the other regions. In terms of electricity generation, while the share of the OECD countries is expected to fall from 60% in the year 1997 to 48% in the year 2020, that of developing Asia is proposed to grow from 17% to 28%, the world's highest growth rate during the same period [1]. Electricity consumption is projected to grow along with GDP growth at nearly 5% on an annual average. The decisions made in the first 20 years of the 21st century will have great significance for Asia and for the world.

In Southeast Asia, where a large number of countries have different fuel mixes, the natural resources are precious not only as a source of domestic energy but also as a source of foreign currency. However, Thailand and the Philippines consume more energy than can be produced domestically (Fig. 1). In terms of emissions, some countries, such as Indonesia and Thailand, produce much more CO_2 per energy unit than that of the world average. The level of CO_2 emissions will rise along with their economic growth without countermeasures against the emissions.

Nuclear energy now faces severe economic competition with other energy sources: coal and natural gas. In general, this competition is caused by the present low prices of fossil fuels and by the continuous development of thermal power plant technology. However, other concerns, such as the disposal of radioactive wastes, nuclear weapons proliferation issues, high capital costs and opposition from local populations, must be solved before nuclear energy can resume its growth.

Half of the nuclear power plants that started operation in the 1990s are located in Asia, including Japan, South Korea, Taiwan, India, Pakistan, and China. In Southeast Asia, no country has yet decided to use nuclear energy although Vietnam appears more interested in it than the others [2]. For a successful introduction of nuclear power plants, the maturity of the market in each nation should be considered in the energy policies as well as economic levels, education standards, environmental concerns and energy security issues. Furthermore, cooperation and assistance from major nuclear powered countries in the fields of manufacturing, engineering, construction, operation, management, and financing will be required. Some of the Southeast Asian countries have already reached a high GDP level per unit of electricity use, similar to the levels at which Japan, South Korea and Taiwan introduced commercial nuclear power plants (Fig. 1), and these levels will further grow in the region.

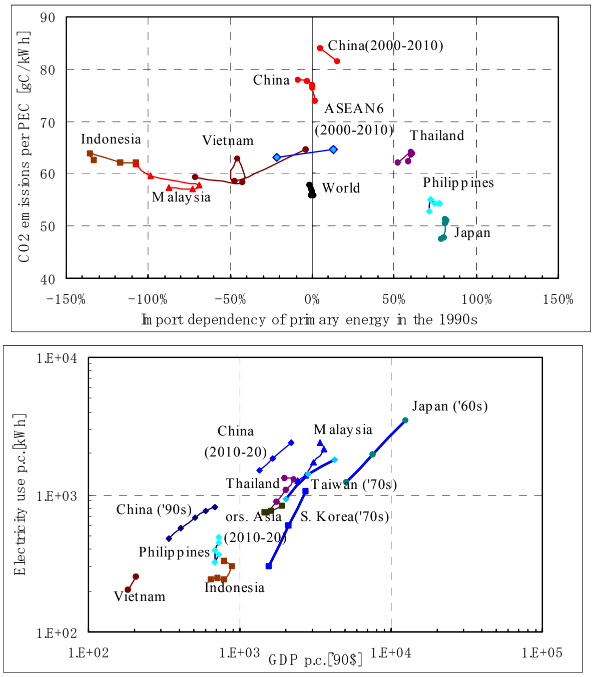


FIG. 1. Primary energy dependency vs. CO_2 emissions, and Trajectory of GDP vs. kWh [3, 4,5].

The objective of this study is to assess the effectiveness of nuclear energy under various circumstances and constraints. In order to do this, an electric power system model, which incorporates economical and technological factors surrounding a country, will be used to determine the performance of energy resources in the future. Minimizing electricity costs as well as environmental burdens at the same time will be ultimate goals for utilities. Those are simulated using the model, for example, in Thailand and Vietnam.

2. CASE STUDIES FOR SOUTHEAST ASIA

2.1. Electric power system model

Electric power system models can be used for many objectives: to determine the optimal mix of energy sources; the optimal distribution of electricity network or the feasibility of introduction of new plants. The main features of the model in this study are as follows:

(a) Incorporating the future projections of electricity demand and fuel prices during a given period,

(b) Comparing levelised costs to decide what kinds of power plants are introduced while fitting the shape of load duration curves,

(c) Knowing the priorities of improvements for a generating technology to become competitive,

(d) Categorizing expansion plans only by fuel type regardless of location, utilities, and so forth,

(e) Considering the uncertainties for electricity demand growth, fuel price escalations, and demand load shape, using approximate probabilistic distributions.

The simulation is constituted mainly of four parts: (1) electricity demand projections; (2) cost projections; (3) expansion planning; and (4) operation (Fig. 2). The planning period covers the first twenty years of the 21st century, from the year 2000 through the year 2020.

3. CRITERIA FOR THE ELECTRIC POWER SYSTEM

(a) Reliability

Reliability is the most important design criterion of electric power. In order to maintain the desired level of reliability and ensure against shortages, some reserve margin must be maintained. The loss-of-load probability (LOLP) or loss-of-energy probability (LOEP) is used to evaluate the reliability as well.

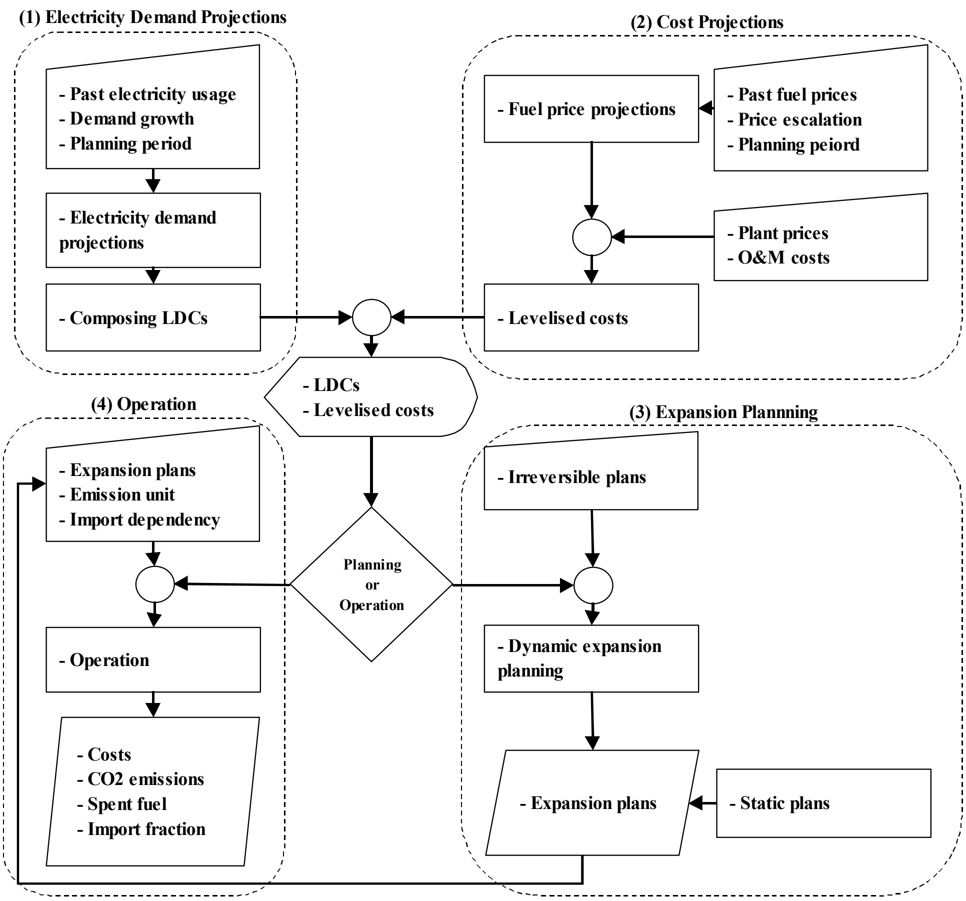


FIG. 2. Flowchart for the simulation.

(b) Costs

The goal of electric utilities is to supply reliable electricity at low costs. Operating and capital costs are major issues in planning an electric power system. The major part of capital costs is the overnight construction cost. Even if the same kind of power plant is constructed, the construction cost greatly depends on the location of sites, financial structure, a cash flow scheme, and its lead-time. Levelised electricity costs that provide useful information for investment choices among different plant options can be used for cost comparisons in equivalent usage conditions. In this study, only construction costs of new plants are considered because the loan payments for existing power plants are made in any expansion plans, unless the loan is paid up or the plants are disposed of.

(c) Environmental concerns

Through the process of generating electricity, many kinds of byproducts, such as ash, SO_x, NO_x and CO_2, are produced (Table I). Much attention is now paid to CO_2 emissions due to the concerns of climate change. In the case of nuclear power, several kinds of radioactive wastes, such as uranium mining, trans-uranium, low-level and high-level radioactive wastes, are produced depending on fuel burn-up and its back-end policies.

(d) Energy securities

On considering energy mix in a region, we must consider an energy option is robust against unexpected disturbances not only for short term but also for long term. In this study, energy security indices for all kinds of fuel are classified into five categories depending on domestic production, consumption, reserves and imports (Table II). With regard to nuclear fuel, an index of 0.5 is taken because uranium ore can be imported from politically stable countries and nuclear fuel will be processed domestically. Also, once the fuel is loaded in reactors, it stays for approximately three to four fuel cycles. So, the prices will be less likely to fluctuate. The indices for hydropower and geothermal power are zero because they are totally indigenous energy sources.

TABLE I. CO_2 EMISSIONS BY GENERATION [6]

Fuel	**[kgC/kWh]**
Coal	0.27
Oil	0.20
Gas	0.178
Hydro	0.005
Nuclear	0.005

TABLE II. IMPORT DEPENDENCY INDICES

Index	**Description**
0	Totally indigenous
0.25	Mostly indigenous but some fraction is imported
0.5	About half is imported
0.75	Mostly imported or dependent on import in the near future
1	Totally dependent on import

4. ELECTRICITY DEMAND PROJECTIONS

Electricity demand projections are derived from several considerations, such as economic growth, population growth, fuel prices, industrial structure, life style, and policies regarding the climate change. For convenience, it is assumed that annual growth rates for both total capacity and generation will be at 4% as a medium case. The maximum demand is assumed to be 85% of the total capacity in the Southeast Asian case, instead of projecting the demand directly. The range of uncertainties takes into account the past records of the growth rates year by year, and the uncertainties are propagated through the planning period. The variances of projections are rather large in Southeast Asia due to the past fluctuating records of electricity demand.

The maximum capacity of nuclear power is set individually, 8GW for the Thai case and 3GW for the Vietnamese case which will occupy 17% and 25% of the total capacity in the year 2020, respectively. Hydropower is precious because its operating costs are the lowest; however, its development depends on particular sites. So, an additional capacity of 2% annually is assumed. Due to the variable availability of water resources, hydropower is assumed to generate, on the average, at a load factor of 30% every year. In today's trends in the generating sector, few if any new oil-fired power plants are expected, which is the same in this simulation. The plants retired during this period are relatively small; thus, they are ignored in this study.

5. LOAD DURATION CURVES

The use of electricity in a region during a given period is expressed using load duration curves (LDCs) as shown in Fig. 3. The LDC is often used in the analysis of an electric power system because it provides information about not only a maximum demand and total generation but also loading characteristics during a period. The LDC can be divided into three parts, such as the base, middle, and peak load range.

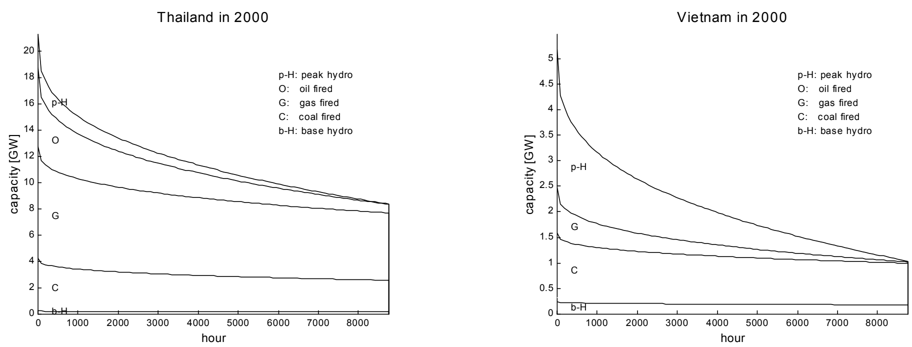


FIG. 3. Load duration curves.

6. COST PROJECTIONS

6.1. Fuel prices

It is difficult to predict fuel prices a decade ahead or so because the prices depend on the reserves of natural resources and political factors in the world. As experienced during the oil crises in 1973 and 1979, the prices of fossil fuels are volatile. Therefore, uncertainties in future prices are incorporated using the same method as the demand projections. It is assumed that

50% of nuclear fuel costs, which are related to the front-end and back-end cycle charges, are so technically oriented that the costs are stable.

For simulation purposes, the fuel prices in the region are assumed to be three quarters of those in Japan when the fuel is indigenous or the same if the fuel is imported (Table III). Fuel price escalations are set as shown in Table IV, such as the base, high and low cases.

TABLE III. FUEL PRICES IN SOUTHEAST ASIAN CASE [7, 8]

Technology	**Coal-fired**	**Oil-fired**	**Gas-fired**	**Nuclear**
Fuel	Steam coal	Heavy oil	LNG	LWR fuel
Price[96$/MWth]	(10×0.75)	(16×0.75)	(14×0.75)	(16×0.75)[$/MWeh]
Imp. dep. (Thailand)	0.5	1.0	0.75	0.5
Index(Vietnam)	0.5	0.75	0.5	0.5

TABLE IV. FUEL PRICE ESCALATIONS

Medium (base case)	Low	High
coal oil gas nuc	coal oil gas nuc	coal oil gas nuc
Price inc.(%) 0.5 1.0 1.0 0.0	0.0 0.0 0.0 0.0	1.0 2.0 2.0 0.5

7. NEW TECHNOLOGY FOR POWER PLANTS

The minimum increment of the generator is taken to be 100MW for coal-fired and gas-fired power plants, and 500MW for nuclear power plants. The maximum load factors of nuclear power plants are set to be 80% reflecting the recent good records. For thermal power plants, the maximum load factors of 70% are assumed based on the current practice, although the latest combined-gas-cycle power plants will be possible to operate at higher levels. The O&M costs are set at three quarters of those in Japan because of the lower prices in this region (Table V).

TABLE V. COSTS AND PERFORMANCE OF NEW TECHNOLOGY FOR THE CASE STUDY [8, 9]

Technology	Capital [$/kW]	O&M [$/kW/yr]	Min. incr. [MW]	Th. eff [%]	Life [year]	Max. LF [%]
Coal	1,000	60	100	35	40	70
Gas	450	37.5	100	45	20	70
Nuclear	2,000	75	500	33	40	80

8. EXPANSION PLANNING

In order to keep the growth of peak load demand, utilities have to add adequate capacity while minimizing costs and environmental burdens under given situations. For short-term planning, marginal cost comparisons take place, as time horizons do not allow the additions of capacity. However, for middle and long-term planning, comprehensive methods should be adopted with changing the decision criteria depending on the economical conditions as well as regulated constraints at any time.

In this study, similar to operation, the optimization of expansion planning is also conducted according to the criteria to realize objectives such as minimizing total costs or environmental burdens. The dynamic expansion planning strategy involves what kinds of plant should be added, watching an average load factor that is expected for the plant under a given LDC. By checking the levelised costs at the load factor among the alternatives, a plant with the lowest levelised costs is selected. This procedure is repeated until the capacity increment meets the requirements for demand growth.

9. DYNAMIC EXPANSION PLANS

Future scenarios in the simulation are set as the combinations of various discount rates and fuel price escalations. Although some scenarios are unlikely, they are useful in the simulation. As a result of using six scenarios, the dynamic expansion planning strategy produced following expansion plans as shown in Table VI. Based on the given assumptions, gas-fired power plants are the most favorable except for the scenario of the low discount rate with the high fuel prices. In Vietnam's "d5-high" scenario, coal-fired power plants are not developed after reaching the capacity limit of nuclear power, which differs from the Thai case, where hydropower will still supply the base load demand.

TABLE VI. DYNAMIC EXPANSION PLANS

Scenario (disc.-fuel price)	Thai case	Vietnamese case
d5-med	gas, then nuclear increment (in 2018)	gas, then nuclear increment (in 2018)
d5-low	gas-only increment	gas-only increment
d5-high	gas, then nuclear (in 2008)	gas, then nuclear (in 2008)
d10-med	then coal increment (in 2014)	then gas increment (in 2017)
low, & high	gas-only increment	gas-only increment

10. THAILAND'S CASE

(a) Operation and its evaluation

In addition to the dynamically composed plans (indicated by bold letters), the other two plans are made as static expansion plans:

Plan 1 : gas-only increment (d10-med, low, high, and d5-low),
Plan 2 : coal and gas increment (equally developed),
Plan 3 : coal, gas and nuclear increment (equally developed),
Plan 4 : gas, then nuclear, then coal increment (d5-high).

The expansion plans are simulated in operation, using the merit order loading procedure. In Plan 4, before introducing nuclear power, coal-fired and gas-fired power plants are operated at their rated load factors because existing oil-fired power plants play the role of peak load supply (Fig. 4). After the introduction of nuclear power plants assigned to the base load supply, the load factor of gas-fired power plants first falls below the rated load factor because the loading order of gas-fired power plants follows coal-fired and nuclear power plants.

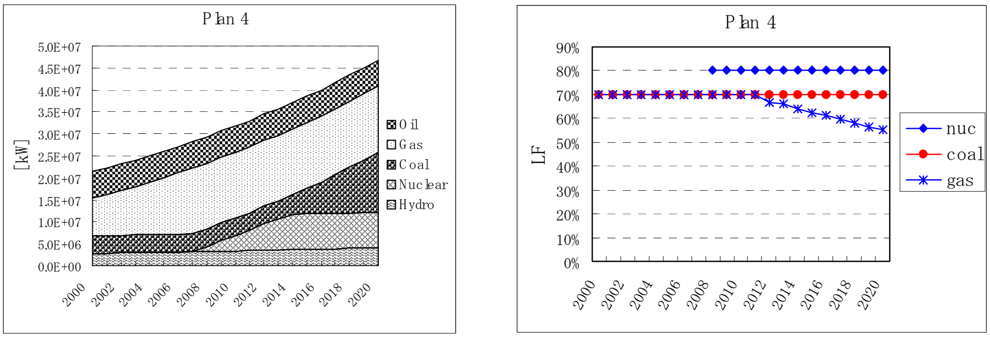


FIG. 4. Generating capacity by fuel type, and Average load factors in Plan 4.

(b) Results

In order to compare the expansion plans, we calculate total costs, CO_2 emissions, produced spent fuel, and imported fuel dependency throughout the planning period.

$$C_{total} = \sum_{i=2000}^{2020} \sum_k C_i^k, \qquad W_{total} = \sum_{i=2000}^{2020} \sum_k \left(W_i'^k \times E_i^k \right), \qquad D_{total} = \sum_{i=2000}^{2020} \sum_k \left(D_i'^k \times E_i^k \right)$$

where C^k_i is annual generating costs of fuel type k, W'^k_i refers to Table I for CO_2 emissions, and $(W'^{nuc}_i)^{-1} = \eta_{heat} \times 35$[GWD/tU] is used for spent fuel calculation. In terms of imported fuel dependency, D'^k_i comes from Table II.

On the annual average, about 30[mill. tC] of CO_2 for the plans without nuclear power plants, and about 26[mill. tC] and 100[tU] of spent fuel for plans with nuclear power will be produced. The minimum costs are realized using the dynamically composed plans in the given scenarios, although in a few cases earlier introduction of nuclear power plants may realize lower total costs than that using the dynamically composed plans. Total emissions were dependent on the plans, which means loading characteristics were not changed by the scenarios.

The results obtained using six scenarios are also converted into values per kWh: (1) costs per kWh, (2) CO_2 emissions per kWh, (3) spent fuel per kWh, and (4) imported fuel dependency per kWh (Table VII). Here, each value is divided by the median values. The variance of costs per kWh among the plans is rather small in the given scenario from the standpoint of accumulated total cost. In terms of standard deviations (STD) of the costs, gas-fired intensive plans tend to have large STDs due to gas price fluctuations. In particular, Plan 3, in which all kinds of power plants are well developed, is stable against electricity demand and fuel price fluctuations. The values of the CO_2 emissions per kWh are nearly twice as much as those in Japan in the year 2000. Gas-fired intensive plans will improve this indicator by 2020 a little, though it will be difficult to realize the significant improvement of this indicator without nuclear power. Spent fuel production per kWh using this level of nuclear power capacity will be almost the same as that in Japan. Thailand is already an energy importing country. Particularly, in Plans 3 and 4, which include the gas-coal-nuclear energy mix, improvements of the imported fuel dependency are observed.

TABLE VII. SIMULATION RESULTS IN THAI CASE UNDER THE "D10-HIGH" SCENARIO

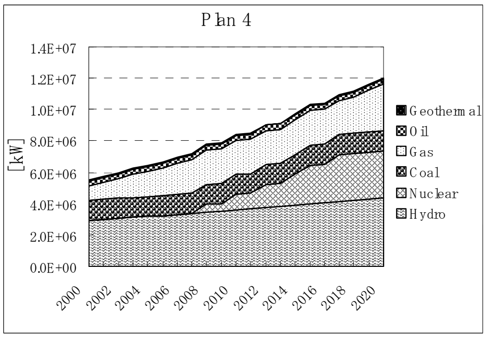
	Cost (STD)	CO_2	SF	Imp. dep.
In 2000	81%	109%	0%	111%
Plan 1	112% (2.9%)	104%	0%	108%
Plan 2	114% (2.1%)	115%	0%	101%
Plan 3	116% (1.9%)	96%	200%	97%
Plan 4	115% (2.0%)	90%	219%	99%

11. VIETNAM'S CASE

(a) Operation and its evaluation

Plan 1 : gas-only increment (d10-med, low, high, and d5-low),
Plan 2 : coal and gas increment (equally developed),
Plan 3 : coal, gas and nuclear increment (equally developed),
Plan 4 : gas, then nuclear, then gas increment (d5-high).

The surging of the average load factors is higher due to the relatively small total capacity compared to the minimum increment unit of power plants, such as 500MW for nuclear power plants (Fig. 5). The load factors of fossil-fired power plants and nuclear power plants will be affected by the contribution of hydropower, which still occupies a significant portion of the total capacity in this country. As a result, newly added gas-fired plants in particular and also coal-fired power plants may operate below their rated load factors, depending on the available water resources in any year.



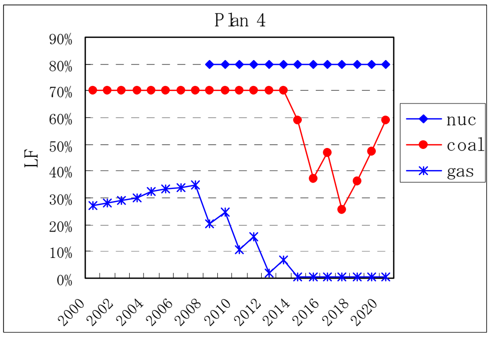


FIG. 5. Generating capacity by fuel type, and Average load factors in Plan 4.

(b) Results

On the annual average, about 4[mill. tC] of CO_2 for the plans without nuclear power plants, and about 2.5[mill. tC] and about 30[tU] of spent fuel for the plans with nuclear power will be produced. The effect on the reduction of CO_2 emissions is significant in the plans including nuclear power plants.

As shown in Table VIII, Plans 3 and 4, which include nuclear power, are divided into two groups in terms of costs depending on the interest rates. Once the capital costs for power plant projects are obtained with good conditions, the projects will be robust throughout the period. The relatively high fraction of hydropower will contribute to stabilizing the costs of the projects. In the year 2000, CO_2 per kWh is higher than that of Japan due to the low availability of hydropower and the small usage of natural gas. However, by using 3 GW of nuclear power, CO_2 emissions per year are, on average, controlled quite well although this merit is counter-balanced with spent nuclear fuel production, as well as with costs in the scenarios with the high interest rate. Vietnam is now providing fossil fuels for themselves, in addition to hydropower. Along with the increase of energy usage, imported fuel dependency may rise. In the simulation, dynamic expansion planning strategy produces some extra capacity due to the assumed minimum incremental capacity of the plants. Thus, introducing the base load plants as soon as possible will reduce the LOEP.

TABLE VIII. SIMULATION RESULTS IN VIETNAMESE CASE UNDER THE "D10-HIGH" AND THE "D5-HIGH" SCENARIOS

	Cost (d10-high) (STD) (d5-high))	CO_2	SF	Imp. dep.	(Cost)
In 2000	45%	105%	0%	106%	
Plan 1	113% (2.5%)	107%	0%	100%	(100%)
Plan 2	116% (1.6%)	124%	0%	100%	(100%)
Plan 3	119% (1.7%)	93%	200%	100%	(98%)
Plan 4	120% (2.1%)	65%	326%	100%	(97%)

12. SENSITIVITY ANALYSIS

12.1. Improvements of nuclear power plant's performance

A high discount rate is likely in the Southeast Asian cases due to the relatively high risks related to electric power projects. Under this circumstance, it may be difficult for capital intensive technology, such as nuclear power, to be competitive with natural gas-fired power plants.

In Table IX designated as C1 and C3, nuclear power can be economically feasible using the dynamic expansion planning strategy by the end of the planning period under the "d10-high" scenario in Thai case (in which nuclear power was originally feasible in 2024).

TABLE IX. SENSITIVITY ANALYSIS OF NUCLEAR POWER PLANT'S PERFORMANCE

C1 (in 2017)	Only capital costs are reduced by 15% to $1,700 per kW
C2 (in 2021)	Capital costs, $1,700 per kW, are recovered in 20 years (also in 20 years for coal, and 15 years for gas)
C3 (in 2012)	In addition to the "C2," O&M costs down to $70/kW/yr, availability up to 90%, burn-up up to 45GWD/tU, thermal efficiency up to 40% (also 40% for coal)

13. COMPARISON OF UNIT SIZES

The size of power units is determined considering technical as well as economical factors. Small sized reactors may be desired in most developing countries due to their small and disconnected electricity grids. At present, reactors of less than approximately 600MW are not available yet. However, in order to stabilize an electricity grid, a single generating unit should be, for example, less than 10% of the total capacity. As an example, operation using the expansion plan obtained by the reactor size of 200MW in Vietnamese case is shown in Fig. 6. In comparison with Fig. 5, the smooth transition of load factors of fossil-fired power plants will contribute to reserving their fuel consumption and its costs.

Moreover, in order to use the scale merit and the multi-unit capabilities of a single site, even more electricity demand is desired. In general, the capital costs of a plant increase less rapidly than its capacity [10]:

$$\frac{\text{(capital costs of large unit)}}{\text{(capital costs of small unit)}} = \frac{\text{(capacity of large unit)}^p}{\text{(capacity of small unit)}^p}$$

The exponent p is normally less than 1, and will be in the range of 0.5 to 0.9 depending on the relative size of the units. However, small sized reactors have advantages, such as learning effects because of many additions, factory fabrication, fast construction (short lead-time), and simplicity for safety features.

$$\text{cost of } n \text{ th unit} = \text{(cost of 1st unit)} \times n^c$$

where $c = \ln L / \ln 2$, and L is the learning coefficient which is determined by the rate of cost reduction. If the unit cost reduces by 10% for each doubling of cumulative production, $L = 0.9$.

Taking into account the above, the dynamic expansion planning is carried out in Thai case and Vietnamese case using the "C3" for 500MW sized reactor as a base. We postulate p=0.9, and L=0.95 or 0.90 for large or medium/small sized reactors, respectively. The learning effects work when plants are introduced in different years rather than in every unit. As shown in Table X, capital costs are various dependent on grid sizes. The number of times for plant introduction is shown in the parentheses.

TABLE X. CAPITAL COSTS WITH SCALE MERIT AND LEARNING EFFECTS

(Reactor size)	1,000MW	500MW	200MW
Thai case	\$1,475/kW(6)	\$1,571/kW(6)	\$1,581/kW(6)
Vietnamese case	\$1,518/kW(3)	\$1,569/kW(6)	\$1,508/kW(9)

Project finance in developing countries is characterized by various forms of investment scheme, diverse participants including governments, and a distribution of benefits and risks. Fig. 7 shows capital needs, revenue and Return on Investment (ROI) using the "C3" with the scale merit and the learning effects of various unit sizes in Vietnamese case. Here, electricity prices are set same as the highest generating cost among all operating power plants at a certain point of the LDC.

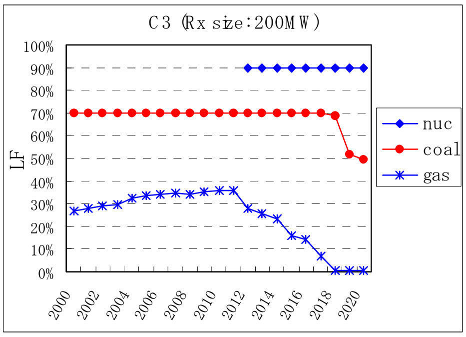


FIG. 6. Average load factors.

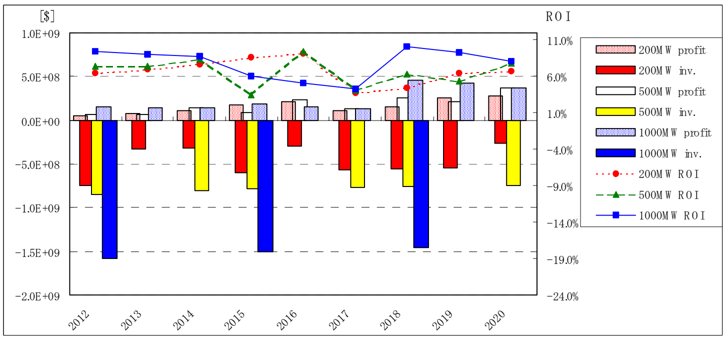


FIG. 7. Balance of the projects.

CONCLUSIONS

A general exploration of the potential for nuclear energy in Southeast Asia was obtained through surveys and simulations in this study. The following conclusions are reached:

(a) A growing economy will create high demand for electricity in the early part of the 21st century. The selection of power plants greatly affects not only the economy but also the environment and energy security. Nuclear energy can play an important role in the best energy mix in this region, particularly as a stabilizing factor against economic disruptions due to unexpected fossil fuel price escalations, global CO_2 emission concerns, and energy security vulnerability.

(b) For the countries addressed as case studies, the results were variable depending on the future scenarios and adopted expansion plans. Gas-fired power plants are the most suitable in many scenarios in terms of costs. However, in some scenarios the profit margin of gas-fired power plants over nuclear power might decrease along with the price escalation of natural gas, making nuclear power more competitive. Under the situation of rapidly growing demand for electricity, adding base load facilities selectively will increase the stability of the electric power system technically as well as economically.

(c) In the sensitivity analysis, improvements of nuclear power plant performance were found to affect the timing for its introduction. The reduction of capital costs is the most important factor, although for long term the reduction of O&M costs will also be critical. Particularly, for small grid areas, adding small sized plants in a timely manner will contribute to the reduction of financial burdens as well as grid stability. Asia is so divergent because of large national differences in economic strength, geography, energy policy and regulation that the best-suited size and type of reactors for individual regions should be carefully selected.

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* This report is based on the thesis submitted by the first author to the Department of Nuclear Engineering at M.I.T. in partial fulfillment of the requirements for the degree of Master of Science in June 1999.

ECONOMIC ASSESSMENT OF SMART IN THE REPUBLIC OF KOREA

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Abstract

This study deals with the investigation of the economics of SMART (System-integrated Modular Advanced ReacTor) with 330MWth class aiming at producing both electricity and desalinated water. SMART is currently at the basic design stage and is being developed by KAERI. The target capacity of water production was predetermined to be 40,000m^3/day. The economic competitiveness of electricity generation was analyzed for various power options including SMART. The desalinated water cost was also analyzed for MED by using the Desalination Economic Evaluation Programme (DEEP), which was developed by the IAEA. In the case of using SMART for power only plant, the electricity generation cost of SMART is estimated to be comparable with that of the alternative power options. In particular, it is estimated that SMART can be competitive with gas combined cycle plant. Although the cost data used in the analysis includes many uncertainties, these results show that SMART can be considered as a potential power option.

1. INTRODUCTION

SMART is an integral type pressurized water reactor with a thermal capacity of 330MW. It is in the design stage, being developed for the dual application of electricity generation and seawater desalination.

The process of evaluating the economics of SMART basically belongs to a joint product problem in the classical economic approach, for SMART aims at producing both electricity and potable water. Therefore, the allocation of the costs to the two final products of electricity and water has a great effect on the economic assessment of a dual-purpose plant. This study focuses on the economic evaluation of SMART being regarded as a power only plant in terms of levelized generation costs. This economic evaluation approach is adopted to prevent the uncertainties due to joint product economics from distorting the economic evaluation of SMART. Potential economics of SMART was performed by comparing the levelized generation cost of SMART with those of other power options in Korea.

Construction cost is one of the major input data items having an important influence on the economics of SMART. However, it is difficult to obtain reliable data, since there is no construction experience of small sized reactors in Korea. Therefore, the construction cost was produced based on a rough estimation. A sensitivity analysis was performed with regard to such major parameters as capacity factor, discount rate, and construction cost.

The economic analysis of SMART coupled with the desalination plant was also carried out using the power credit method in the analysis. The power credit approach is an adequate method for evaluating the economics of a power dominant dual plant.

2. CONSTRUCTION COST ESTIMATION

It is difficult to obtain reliable data on the construction cost because SMART is at the basic design stage. Therefore, a simple estimation method of adjusting the scaling factor was applied in this study to calculate construction cost of SMART.

On the other hand, there was a recent study to calculate the construction cost of SMART in Korea. The study was carried out by KOPEC(Korea Power Engineering Co.). KOPEC's study tried to figure out the proper range of construction costs in as much detail as possible based on the design concept [1].

To ensure the reliability, the construction cost produced by a simple estimation method in this study was compared with KOPEC's study.

2.1. Construction cost estimation

The construction cost of SMART was estimated based on the existing 1000MWe PWR data in Korea. Accordingly, it seems that some uncertainties are inevitably involved in the estimation. In this study, overnight cost was calculated by applying a scaling factor of 0.8 to the conventional nuclear power plant of 1000 MWe. Since the overnight cost of PWR with 1000MWe class was 1,541$/kWe referred in Electric System Expansion Planning in Korea, overnight cost of SMART with 100MWe is calculated to be 2,442$/kWe by applying scaling factor of 0.8 as follows:

Overnight cost of SMART = Overnight cost of 1000MW PWR $\times$ (Capacity of PWR/Capacity of SMART) $\times$ (Capacity of SMART/Capacity of PWR)^{p} where, p means a scaling factor.

SMART is simplified by adopting an integral and modular type primary circuit. Owing to this advanced characteristics, the overnight cost of 2,442$/kWe reflecting only economies of scale could be reduced further. In this study, it is assumed that there is 25% or margin to be reduced in the overnight cost calculated above.

Therefore, the overnight cost of SMART is assumed to be 1,800$/kWe as a reference case.

2.2. Construction cost estimation based on design concept

The construction cost estimation was recently carried out by KOPEC. That study was based on the design concept and tried to reflect the design concept to the estimation of construction cost in detail. However, that study also involves a lot of uncertainties, because of there being no detailed design to rely on.

The direct cost consists of reactor plant equipment, turbine plant equipment, civil construction work, and electric & machinery plant equipment.

The reactor plant equipment cost is based on the estimated values of DHIC(Doosan Heavy Industries & Construction Co., Ltd) reflecting the construction experience of KSNP(Korean Standard Nuclear Plant).
Turbine plant equipment cost is estimated based on international prices reflecting the construction experience of KSNP. The costs of civil construction work and of electric & machinery plant equipment are based on the construction experiences of KSNP.

According to KOPEC's study, an overnight cost of First-Of-A-Kind plant appears to be 2,674$/kWe. The study applied the appropriate learning curve factors to calculate the costs for each subsequent unit. The Overnight cost of the fourth unit is calculated to be 2,213$/kWe. In addition, the study shows that it can be reduced further to 1900$/kWe reflecting the learning effect from subsequent construction.

COMPARATIVE ECONOMIC ASSESSMENT FOR POWER OPTIONS

The main objective of this analysis is to figure out the potential economic viability of SMART by comparing the economic competitiveness of alternative power options. In this respects, levelized electricity generation costs are calculated for coal fired, gas combined cycle, conventional nuclear, and SMART, and the results are compared.

2.3. Major input data

Major input data are given in Table 1. The output scales of power options are referred to an existing Electric System Expansion Planning. The output of a coal fired power plant is assumed to be 500MWe, gas combined cycle is 450MWe, conventional nuclear is 1,000MWe, and SMART is 100MWe. The life time for conventional nuclear and SMART is assumed to be 40 years, while for coal fired power plant and gas combined cycle 30 years is assumed. The cost data for construction, O&M, and fuel for all the power options except SMART were referred to the input data of an existing Electric System Expansion Planning in Korea. Capacity factors are assumed to be 80% for all power options. Annual Discount rate of 8% was applied.

TABLE I. MAJOR INPUT DATA

	Unit	Coal fired	Gas combined cycle	Conventional Nuclear	SMART
Capacity	MWe	500	450	1000	100
Life time	Year	30	30	40	40
Overnight cost	US$/kWe	1,043	520	1,541	1,800
O&M cost	US$/kWh	0.006	0.004	0.008	0.008
Fuel cost	US$/kWh	0.012	0.030	0.004	0.008
Capacity factor	%	80	80	80	80
Discount rate	%	8	8	8	8

TABLE II. LEVELIZED ELECTRICITY GENERATION COST FOR POWER OPTIONS

Unit:$/kWh

	Coal fired	Gas Combined Cycle	Conventional Nuclear	SMART
Capital Cost	0.013	0.007	0.018	0.024
O&M Cost	0.006	0.004	0.008	0.008
Fuel Cost	0.013	0.031	0.004	0.008
Generation Cost	0.032	0.042	0.030	0.040

2.4. Levelized electricity generation cost for power options

The levelized electricity generation costs given in Table 2 show that a conventional nuclear system is the most economical power option, followed by a coal fired plant and then SMART. Gas combined cycle appears to be the most inferior power option in the economic assessment comparison. The levelized generation cost of SMART seems to be 33% higher than that of conventional nuclear, the most competitive power option, and 25% higher than that of coal fired power plant.

3. SENSITIVITY ANALYSIS

Capacity factor and discount rate were identified as the parameters having the largest impact on the levelized electricity generation costs of the power options. So, sensitivity analyses with respect to these two parameters were carried out.

3.1. Sensitivity Analysis with respect to Capacity Factor

The results of sensitivity analysis with respect to the capacity factor were given in Table 3. It can be seen from Table 3 that the comparative economics of gas combined cycle is improved as capacity factor goes down. A capacity of 30% will make gas combined cycle the most economic option. On the other hand, as the capacity factor increases, the comparative economics of conventional nuclear power is improved so that a capacity factor of 70% will make conventional nuclear, together with coal fired power plant, more economical than other power options. For capacity factors higher than 70%, conventional nuclear would be the most economical option. As for SMART, it seems to be competitive with gas combined cycle depending on capacity factors considered here. As the capacity factor increases, the comparative economics of SMART over gas combined cycle seems to improve, as expected.

TABLE III. SENSITIVITY ANALYSIS WITH RESPECT TO CAPACITY FACTORS

Unit: $/kWh

Capacity factors	Coal fired	Gas Combined Cycle	Conventional Nuclear	SMART
30%	0.064	0.060	0.074	0.062
50%	0.043	0.048	0.046	0.051
70%	0.034	0.043	0.034	0.043
80%	0.032	0.042	0.030	0.040
90%	0.029	0.040	0.027	0.038

3.2. Sensitivity Analysis with respect to Discount Rate

Table 4 shows the sensitivities of levelized electricity generation costs to variations of discount rates. Higher discount rates favour the power options with lower construction cost, while lower discount rates favour the power options with higher construction cost. For the discount rate of 10%, nuclear, together with coal fired plant, appears to be more economical than other options. A discount rate of 12% will make a coal fired power plant more competitive than a nuclear one. As for the comparison between SMART and gas combined cycle, the gas combined cycle is more economic for the discount rates higher than 10%. However, it appears that SMART is more economical than a gas combined cycle at the discount rate lower than 8%. At the discount rate of 8%, economic superiority of SMART over gas combined cycle seems to be 5% in terms of levelized electricity generation unit cost.

TABLE IV. SENSITIVITY ANALYSIS WITH RESPECT TO DISCOUNT RATES

Unit: $/kWh

Discount rate	Coal fired	Gas Combined Cycle	Conventional Nuclear	SMART
3%	0.026	0.039	0.021	0.028
5%	0.028	0.040	0.025	0.032
8%	0.032	0.042	0.030	0.040
10%	0.034	0.043	0.034	0.046
12%	0.037	0.044	0.039	0.052

4. LEVELIZED ELECTRICITY GENERATION COST OF SMART

The impacts of the major parameters such as overnight cost, capacity factor, and discount rate on the levelized electricity generation cost of SMART were investigated, because these parameters have been identified to have a large influence on the competitiveness of SMART. The economic parameters adopted in the reference case are as follows: discount rate is 8%, capacity factor is 90%, and overnight cost is 1,800$/kWe. A relatively high capacity factor of 90% is considered in SMART, because its design characteristic compared to the conventional power plants leads to low outage rates.

4.1. Sensitivity Analysis with respect to Overnight Cost

Three cases for overnight costs are investigated to figure out the influences of the construction costs on levelized electricity generation cost of SMART. They are assumed to be 1,600$/kWe, 1,800$/kWe, 2,000$/kWe respectively.

According to the sensitivity analyses with respect to overnight costs, levelized electricity generation cost was estimated to be 0.038$/kWh, 0.040$/kWh, and 0.043 $/kWh for the assumed overnight costs of 1,600$/kWe, 1,800$/kWe, 2,000$/kWe respectively.

4.2. Combined variation effects of both capacity factors and overnight costs on the levelized electricity generation cost of SMART

Table 5 summarizes the results of the levelized electricity generation cost of SMART, which are calculated according to possible changes of both the overnight cost and capacity factors.

TABLE V. LEVELIZED ELECTRICITY GENERATION COST OF SMART ACCORDING TO THE VARIATION OF BOTH CAPACITY FACTORS AND CONSTRUCTION COSTS

Unit: $/kWh

Overnight Cost Capacity factor	2,000($/kWe)	1,800($/kWe)	1,600($/kWe)
30(%)	0.067	0.062	0.057
50(%)	0.054	0.051	0.047
70(%)	0.046	0.043	0.040
80(%)	0.043	0.040	0.038
90(%)	0.040	0.038	0.035

These analyses were carried out in order to capture the combined variation effects of both capacity factors and overnight costs. The capacity factors are allowed to vary from 30% to 90% for three alternative overnight costs of SMART, which was assumed above. The results show that the levelized electricity generation costs of SMART ranged from 0.035$/kWh to 0.067$/kWh.

4.3. Combined variation effects of both discount rates and overnight costs on the levelized electricity generation cost of SMART

The discount rates are allowed to vary from 3% to 12% for the three alternative overnight costs of SMART, which was assumed above. The results were given in Table 6, which shows that the levelized electricity generation costs of SMART ranged from 0.027$/kWh to 0.056$/kWh.

TABLE VI. LEVELIZED ELECTRICITY GENERATION COST OF SMART ACCORDING TO THE VARIATION OF BOTH DISCOUNT RATES AND OVERNIGHT COSTS

Unit: $/kWh

Overnight Cost Discount Rate	2,000($/kWe)	1,800($/kWe)	1,600($/kWe)
3%	0.029	0.028	0.027
5%	0.034	0.032	0.031
8%	0.043	0.040	0.038
10%	0.049	0.046	0.043
12%	0.056	0.052	0.049

5. DESALINATION WATER UNIT COST CALCULATION

The IAEA Desalination Economic Evaluation Programme(DEEP) was used to calculate water production cost[2]. The MED(Multi Effect Distillation) is only considered as a water production process to be coupled with SMART because of its excellent prospects for economic advantage and technology development. The target water production capacity was predetermined to be 40,000m^3/day to meet the requirements of both electricity and water supply for a population of approximately 100,000.

The power credit method was employed for the desalination water unit cost calculation. This method is widely used for an electricity dominant dual plant.

The major input data necessary for the calculation of water production unit cost came from the DEEP computer code. The maximum brine temperature is assumed to be 70°C. The choice of maximum brine temperature has to be determined in designing the plant. The maximum brine temperature plays an important role in determining gain output ratio(GOR), which, has, in turn, an influence on determination of the base unit cost of desalination plant. In reality, the increase of the maximum brine temperature leads to higher value of GOR, which, in turn, results in higher investment cost and therefore, in higher water production costs, because it needs more sophisticated materials for the purpose of achieving the higher plant efficiency brought about by higher GOR.

A discount rate of 8% and a capacity factor of 90% were also assumed in the calculation of desalination water unit cost.

5.1. Combined variation effects of both capacity factors and overnight costs of SMART on water production unit costs

The combined variation effects of both capacity factor and overnight costs of SMART on water production unit costs were investigated. The capacity factors are allowed to vary from 30% to 90% for the three alternative overnight costs of SMART, which was assumed to be 2,000$/kWe, 1,800$/kWe, 1,600$/kWe respectively.

Table 7 summarizes the results of water production unit cost, which are calculated according to possible changes of both capacity factors and the overnight cost of SMART. The results show that water production unit costs ranged from 0.73$/m^3 to 1.18$/m^3.

TABLE VII. COMBINED VARIATION EFFECT OF BOTH CAPACITY FACTORS AND OVERNIGHT COSTS OF SMART ON WATER PRODUCTION UNIT COST.

Unit: $/m^3

Overnight Cost / Capacity factor	2,000($/kWe)	1,800($/kWe)	1,600($/kWe)
30(%)	1.18	1.17	1.15
50(%)	1.05	1.04	1.03
70(%)	0.91	0.90	0.89
80(%)	0.83	0.82	0.81
90(%)	0.75	0.74	0.73

TABLE VIII. COMBINED VARIATION EFFECT OF BOTH DISCOUNT RATES AND OVERNIGHT COSTS OF SMART ON WATER PRODUCTION UNIT COST

Unit: $/m^3

Overnight Cost / Discount Rate	2,000($/kWe)	1,800($/kWe)	1,600($/kWe)
3(%)	0.52	0.52	0.51
5(%)	0.61	0.60	0.59
8(%)	0.75	0.74	0.73
10(%)	0.86	0.85	0.83
12(%)	0.98	0.96	0.94

5.2. Combined variation effects of both discount rate and overnight costs of SMART on water production unit costs

Water production unit costs are calculated according to possible variations of both discount rates and the overnight costs of SMART. In the water production unit cost calculation, discount rates are assumed to vary from 3% to 12% for the three given alternative overnight costs of SMART. Table 8 shows that the water production unit costs ranged from 0.51$/m^3 to 0.98$/m^3.

CONCLUSION

In the case of using SMART as a power-only plant, the result shows that the electricity generation cost of SMART is comparable with that of the alternative power options. In particular, this study shows that SMART can be competitive with gas fired combined plant. This result indicates that SMART can be considered as a potential power option. In addition, an economic analysis of SMART coupled with MED was performed. The result shows that the calculated water production unit costs are in the range of 0.73-0.83($/m^3) assuming plant availability of 80% or higher with a discount rate of 8%. This indicates that SMART can be considered as a competitive choice for desalination.

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ECONOMICS OF ADVANCED LWR AT SMR POWER RANGE IN DEVELOPING COUNTRIES

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Abstract

Recently, an alternative approach of efficient electric power generation consisting of a combined cycle with nuclear and gas thermal power has been presented [1]. A consistent economic analysis of this approaches been developed by Florido et al [2], suggesting the convenience of the gas-nuclear coupling for developing countries.

Contrary to the trends followed from classical assessments of nuclear and gas power generation taken separately [1], the maximization of the superheated temperature was not found to be a good design criterion. In particular in smaller electric grids, CCGTs combine better with SMR than conventional LWR and HWR.

For systematic analysis of integrated SMR on electrical grids, the IAEA developed the *IREP1.0* code. This code makes several approximations, valid in the power range of large reactors, and is not suitable for the assessment of SMR in small grids.

In this article the advanced version of *IREP1.0* is presented. *IREP2.0* is provided with special models suitable for developing countries' grids. Moreover *IREP2.0* includes the capabilities to analyze SMR coupled with gas turbines (GT). Using *IREP2.0*, a consistent economic assessment of integrated LWR for developing countries is performed. Also the combination with GT has been analyzed, suggesting the possibility of lower levelised costs.

The results of this analysis clarify the potential advantages of Integrated SMR for developing countries, the real advantages or not of coupling an Integrated LWR with a CCGT dependent on gas prices, and which is the best design from the economic point of view.

The economic assessment analysis must be done carefully for each country taking into account several parameters concerning geological, political and economical situations. Thus the *DuCom* option is not always the best choice for all situations. These factors will be reflected in the calculation of capital costs, operation & maintenance costs, engineering costs, discount rates, electricity and gas prices.

Many developing countries require low capital investment, which means small and medium power generators. Thus the analysis is focused on small and medium integrated reactors.

1. INTRODUCTION

Small and Medium Reactors(SMR) have been studied as promising alternatives for competitive nuclear energy in small and medium electric grids. However the economic assessment of these new designs should include innovative methods of analysis, accounting for the forecasting of future electric markets, financial risks, dynamics of prices among others.

When natural gas is available at low prices, Combined Cycle Gas Turbines (CCGT) are very strong competitors against nuclear power. One strong advantage of Gas Turbines is the relatively low power of the biggest turbines together with the low economic scale at high power range. Then the flexibility of this type of energy is a very powerful tool for developing countries.

Classical LWR and HWR reactors typically have five to ten times higher power output, with capital requirements more than an order of magnitude higher. Most efforts are presently focused on reducing the capital costs and construction times of nuclear power plants. Although some of these efforts have been directed to maintain the reliability of present water-cooled reactors, it has not been possible to improve the thermal efficiency. On the other hand, alternative coolants and fuels, would enhance the thermal efficiency, but the benefits of the reliability of proven technologies would be lost.

Figure 1 [2] shows the potential position of different countries in a map of competitiveness regions. Gas prices and construction costs of nuclear plants are plotted in a two dimensional plane where the best alternative for power generation can be visualized by zones. The dashed line indicates the competitiveness boundary without taking into account the combined option, i.e. comparing nuclear against gas separately. Interestingly most of the countries fall in the region where the combined nuclear-gas cycle is potentially more competitive.

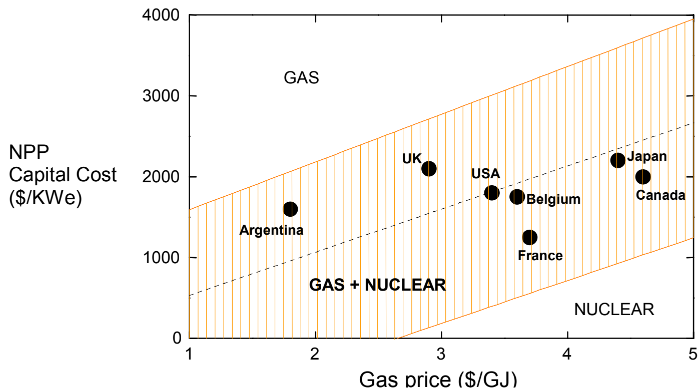

Figure 1. Competitivity map showing the most convenient alternative.

The competitiveness map shown in figure 1 is based on simplified model of superheating of the nuclear secondary circuit steam [2]. The competitiveness between nuclear and gas is determined principally by the gas price. Wherever the gas is expensive, nuclear power plants would be recommended; whereas in regions where gas is available at low prices, CCGT are preferable.

Bearing this in mind, deeper analysis has been made in order to obtain more accurate results. IREP1.0 [4] has been modified, improved and extended for nuclear-gas combined thermal cycle calculation.

2. IREP CODE

IREP code is an *Integrated Reactor Evaluation Program*. The code is primarily designed for low power integrated PWRs, with integrated and self pressurized primary system (Figure 2).

Different engineering solutions has been used for steam generators and circulation of the primary coolant.

- Steam generator cassettes with primary system in the tube side or in the shell side.
- Natural or forced convection for the primary coolant circulation.

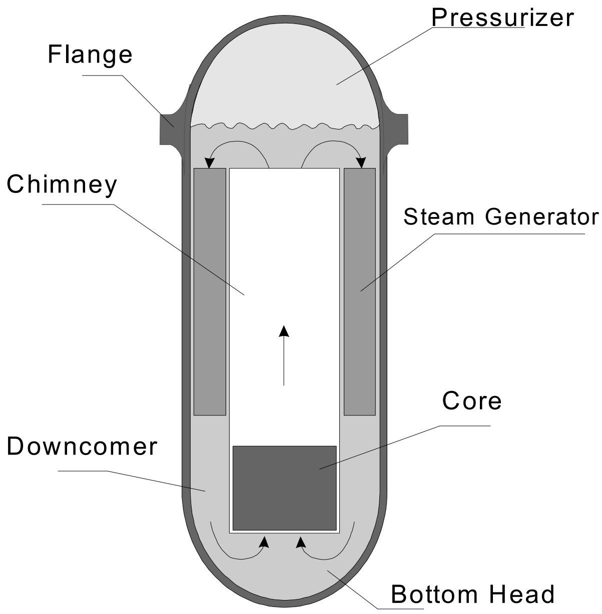

Figure 2: Integrated Reactor Pressure Vessel.

IREP code presents a strong coupling among several variables concerning mechanics, neutronics and thermal-hydraulics calculations. This feature is intimately related to the strong dependence of thermal balance and the lay-out of the reactor components.

2.1. Mechanics

Dimensions, weights, thicknesses and gaps of several parts and components of reactor are calculated. This calculation accounts for ASME criteria and constraints derived from thermal balance calculations.

2.2. Thermal-hydraulics

Pressure drops in the core, riser, steam generators, down-comer, spacers, etc, are calculated using experimental correlations. Power generation in the core and the cooling in the steam generator are balanced with the pump power and frictional losses mentioned before.

One dimensional finite difference methods and several nested iteration loops with over-relaxation techniques are used.

Thermal balance is used between the friction losses, buoyancy force and, power and pressure pump head.

Also maximum heated channel, various power peaking factor, cosine power distribution, mixing factors, etc are considered in order to calculate the critical power ratio.

2.3. Neutronics

The Neutronic model includes, the beginning of life (BOL) reactivity, extraction burn up, and void reactivity coefficient. The batch irradiation is calculated together with the load factor of the plant.

The number of fuel elements in a batch, power density, enrichment, and many others are modifiable design parameters at run-time.

The core radius is calculated automatically, minimizing the radial core leakage and the closest to the linear power generation heat rate.

2.4. Economic

The capital costs are estmated using the classic scaling method. The time distribution of investment defined at run time is used to calculated the net present value of the total capital cost. Also fuel cycle costs, first core amortization and refueling costs are considered.

Then with the load factor, the total unit energy cost (TUEC) in mill$/kWh, is calculated.

2.5. Optimization

One of the remarkable features of the IREP code is its automatic optimization. Using the gradient method applied on multi-dimensional space of TUEC, the IREP code "searches" for the optimal parameters that minimize the TUEC of the plant.

TABLE I: VARIABLES AVAILABLE FOR AUTOMATIC OPTIMIZATION
Primary System Pressure
Primary Pumps Flow
Inlet SG temperature
Secondary System Pressure
Diameter of SG tubes
Length of SG tubes
Fuel Rod Pitch
Enrichment
Gas Turbine Flow Rate

The table I shows 9 variables that define a nine-dimensional space. These variables can be deactivated or activated for the optimization depending on the problem or analysis.

IREP2.0

IREP2.0 code includes a new model of the reactor secondary circuit in order to analyze steam cycles combined with a GT. The code also provides a numerical model for steam turbines allowing two pressure expansion stages. Also an off-design behavior has been studied for the steam turbines, to enable a variable steam flux calculus for load changing.

A heat-exchange calculation code has been developed to find an optimal heat transfer for each nuclear-gas combination. The heat-exchanger model is based on the NTU method.

Figure 3 shows a simplified scheme of the secondary cooling circuit used in the *IREP2.0* code. The heat of the exhausted gas from the gas turbine is transferred to the steam generator outlet, resulting in a higher inlet temperature to the high-pressure steam turbine.

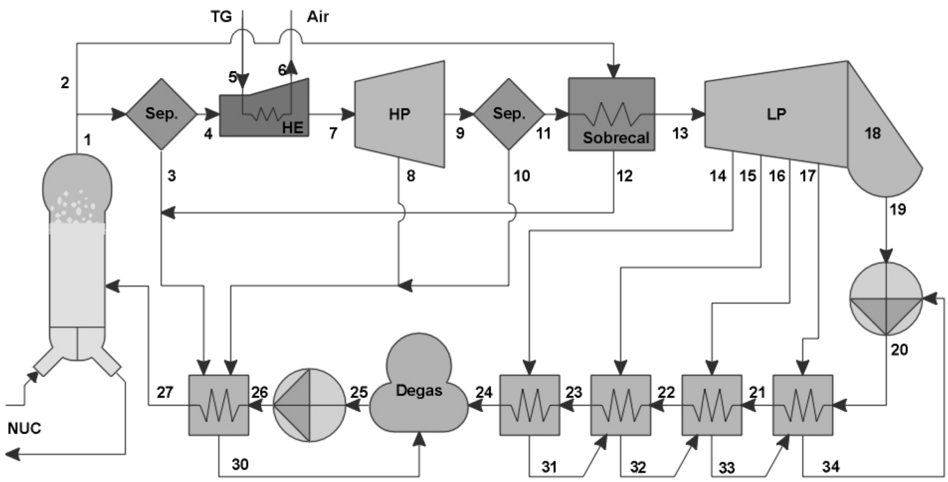

Figure 3: Schematic sketch of IREP v2.0 Secondary Circuit. HE (Heat Exchanger), TG(Gas Turbine), DesGas(DeAirator), SobreCal (Super-heater),Sep (Separator), LP(Low Pressure Steam Turbine), HP (High pressure Steam turbine).

This new approach of combining both sources of energy is called DuCom (DUal-COMbined). It gives a higher thermal efficiency due to the increase in the average temperature of the hot heat source.

IREP2.0 implements user friendly capabilities, such as parametric plots, to facilitate the design feedback analysis. All the mechanical, neutronic and thermal-hydraulic models of primary circuit remain unchanged.

2.6. Performance of DuCom with SMR

The overall cost composition of nuclear-gas combination without synergy, i.e. each source of energy generating electricity independently, is as follows:

$$C = \frac{Pot_{NUC} \cdot C_{NUC} + Pot_{GAS} \cdot C_{GAS}}{Pot_{NUC} + Pot_{GAS}} \qquad (1)$$

Where *Pot and C* indicates electric power generated and cost respectively. Now, the combination of nuclear and gas thermal cycle (DuCom) will produce additional electricity. The source of this power basically, comes from the wasted heat of the exhaust gas from the gas turbine. This additional synergy power will be called Pot_{SIN}.

To obtain Pot_{SIN} it was necessary to install a heat exchanger and there will be additional cost for operation and maintenance. This cost will be called C_{SIN}. Then the new expression for total *DuCom* cost is:

$$C_{DuCom} = \frac{Pot_{NUC} \cdot C_{NUC} + Pot_{SIN} \cdot C_{SIN} + Pot_{GAS} \cdot C_{GAS}}{Pot_{NUC} + Pot_{SIN} + Pot_{GAS}} \quad (2)$$

For almost every situation C_{DuCom} is lower than *C (ec1)*. Nevertheless *DuCom* will not be the best choice unless C_{DuCom} is lower than C_{NUC} and C_{GAS}. If this condition can not be accomplished, the best choice is "pure" nuclear or "pure" gas turbine instead the combination of both.

It worth emphasize that the synergism is always present in *DuCom*, but with some additional cost. To reach the region of competitiveness, it's necessary to evaluate the result of the balance between the costs and benefits of this synergy.

Figures 4 show variation of C_{DuCom} for a nuclear power plant of 100MWth in combination with variable gas turbine power and evaluated for four different gas prices. For cheap gas (Fig4a), the "pure" gas turbine option is always the most competitive. For gas at 4\$/GJ (Fig4b), the choice of DuCom equals the GT option for GT power greater than 60MWe. For 6\$/GJ (Fig4c) *DuCom* is the best option for gas turbines greater than 20MWe. This feature persists until gas prices reach 8\$/GJ, from where the most competitive option becomes the "pure" nuclear power plant (Fig4d).

The conclusion is that DuCom is likely to compete when the electricity generation cost of gas and nuclear are similar (Figure4c). This condition depends on capital cost, operation and maintenance cost, fuel element cost, gas prices and discount rate, and therefore is very sensitive to regional variations.

2.7. Engineering Invariance approach of DuCom

An important condition imposed on the system is that, for a given gas turbine and economically optimal nuclear power plant design, the *DuCom* combination of these two plants does not change any engineering parameters of the nuclear reactor.

This assumption is important, it does not only saves considerable amount of resources during design and construction time, but it means that the GT and nuclear reactor can be operated separately or coupled depending on the availability of each source of energy. This feature increases the system load factor, making it even more competitive.

Eq.2 can be then rewritten as following:

$$C_{DuCom} = f_{NUC} \cdot C_{NUC} + f_{SIN} \cdot C_{SIN} + f_{GAS} \cdot C_{GAS} \quad (3)$$

where

$$f_i = {Pot_i} \Big/ {Pot_{NUC} + Pot_{SIN} + Pot_{GAS}}$$

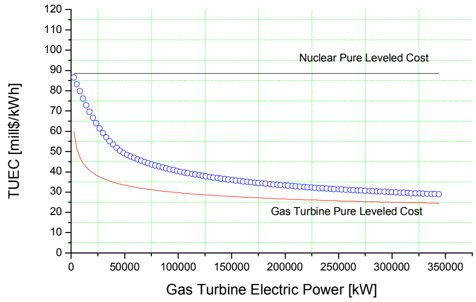

Figure 4a: variable GT power with 1$/GJ of gas price, gas turbine is the best option.

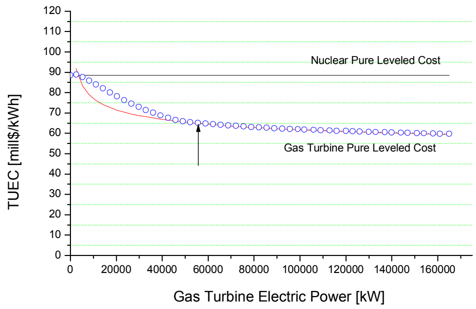

Figure 4b: variable GT power with 4$/GJ of gas price, from 60MWe, the choice of DuCom is indifferent from the choice of gas turbine.

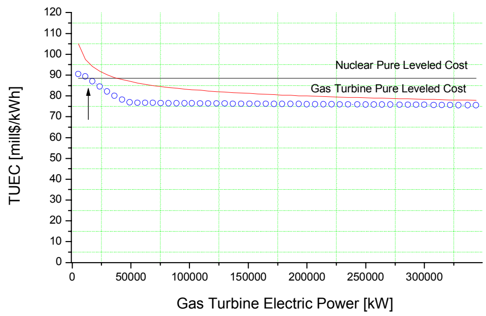

Figure 4c: variable GT power with 6$/GJ of gas price, from 20MWe, DuCom option is the most competitive.

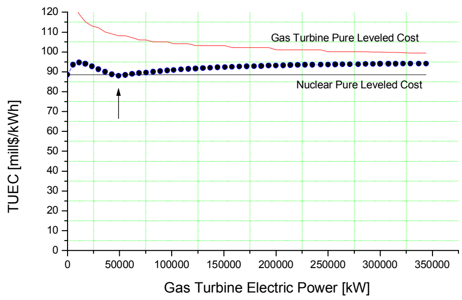

Figure 4d: variable GT power with 8$/GJ of gas price, except for 50MWe the choice of the nuclear energy is the best.

Figure 4: Variation of leveled DuCom cost(circle) for the fixed nuclear plant and variable gas price and gas turbine power. Arrows indicate competitive synergism.

Consider now a reactor, parameter X (for instance, primary pressure, secondary pressure, SG active tube length, etc..). The variation of C_{DuCom} respect to this variable will be:

$$\begin{aligned}\frac{\partial C_{DuCom}}{\partial X} = f_{NUC} \cdot \frac{\partial C_{NUC}}{\partial X} + f_{SIN} \cdot \frac{\partial C_{SIN}}{\partial X} + f_{GAS} \cdot \frac{\partial C_{GAS}}{\partial X} + \\ \frac{\partial f_{NUC}}{\partial X} \cdot C_{NUC} + \frac{\partial f_{SIN}}{\partial X} \cdot C_{SIN} + \frac{\partial f_{GAS}}{\partial X} \cdot C_{GAS}\end{aligned} \qquad (4)$$

If the reactor design is optimum, the first term of the right side of the equation 4 is neglected. The third term also is neglected for X is a reactor parameter and does not affect the gas generation cost.

$$\begin{aligned}\frac{\partial C_{DuCom}}{\partial X} \cong \frac{\partial f_{NUC}}{\partial X} \cdot C_{NUC} + \frac{\partial f_{GAS}}{\partial X} \cdot C_{GAS} + \\ \frac{\partial f_{SIN}}{\partial X} \cdot C_{SIN} + f_{SIN} \cdot \frac{\partial C_{SIN}}{\partial X}\end{aligned} \qquad (5)$$

As stated before to make *DuCom* option competitive, the cost of electricity generation of both (nuclear and gas) source are similar (figure 2c).

$$C_{NUC} \cong C_{GAS} = C$$

and from the definition in equation 3:

$$f_{NUC} = 1 - (f_{SIN} - f_{GAS})$$

replacing these relations in the first and second terms of the eq. 5:

$$\frac{\partial C_{DuCom}}{\partial X} \cong - \frac{\partial f_{SIN}}{\partial X} \cdot C + \frac{\partial f_{SIN}}{\partial X} \cdot C_{SIN} + f_{SIN} \cdot \frac{\partial C_{SIN}}{\partial X} \qquad (6)$$

Eq_6 represents the fraction of the cost-migration *balance* between nuclear-gas and synergism, i.e. net cost variation due to using synergy instead of nuclear and gas.

- $\sigma f_{SIN} / \sigma X$ represents the fractional increment of synergy at the expense of nuclear and gas power.
- The first term represents the cost saving for less power generation at nuclear and gas cost.
- The second and third terms represent the cost of synergy.

For the cases evaluated and analyzed in this work, optimum values have been chosen for the synergism. So that the balance of the Eq 6 leads to zero.

If this assumption is valid, some design parameters of the nuclear power plant could change taking advantage of synergism. For example, primary pressure could be reduced for the cases in which nuclear the superheating-cost is greater than the heat-exchanger-oversizing-cost.

Although full-optimization is desirable from the point of view presented here, there are other important aspects not accounted in the analysis, for example, additional engineering costs of new plant. On the other hand if restricted-optimization is used, -i.e. fixed optimal nuclear-gas plant plus optimal synergy parameters-, some valuable advantages come out, as already mentioned at beginning of this section.

2.8. Analysis

Figure 5 shows the different cost composition between the gas and nuclear power plants. Changes in gas prices will affect to the gas turbine electricity generation cost. While nuclear energy is dominated by capital costs.

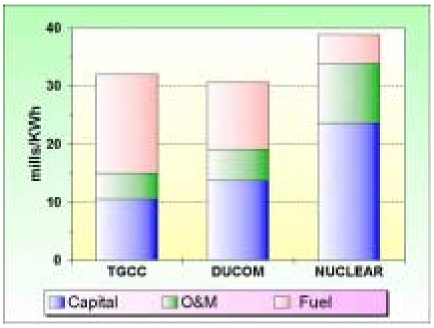

Figure5: Cost composition comparison.

Bearing in mind this cost composition, three main parameters were selected to study their influence on he DuCom economics:

- Nuclear Reactor Power
- Gas Turbine Power
- Gas price

Figure 6 shows iso-cost surfaces of *DuCom* varying in the ranges from 30 to 170 MWe for nuclear power plant, 0 to 200 MWe for gas turbine power and from 2 to 10 $/GJ for gas prices.

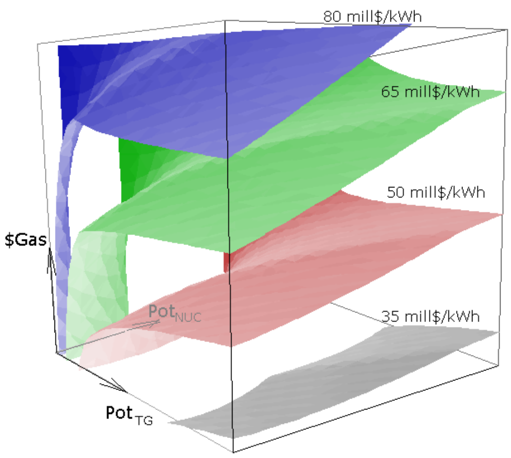

Figure 6: iso-cost surfaces for 35, 50, 65, 80 mill$/kWh.

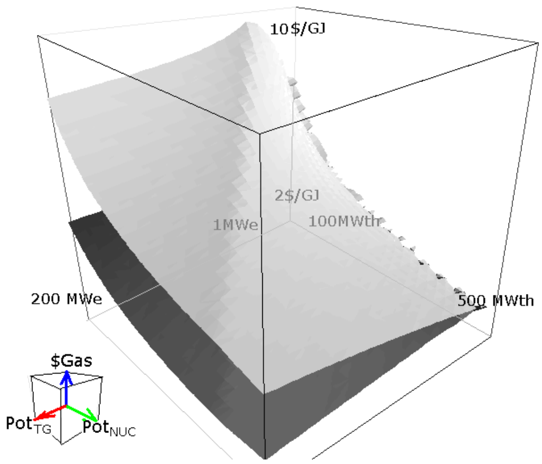

Figure 7: Indifference Surface ($C_{savings}$=0).

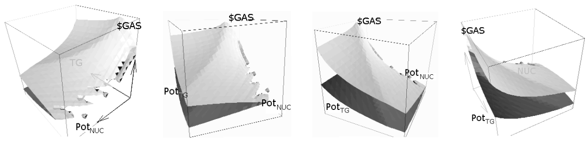

Figure 8: Indifference Surface from different angle of view.

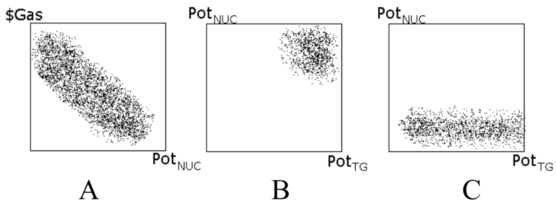

Figure 9: Competitive region plot of DuCom cycle. A) Low gas prices require high nuclear reactor power. B) Low gas prices: reasonable DuCom combination is made for high powers. C) High gas prices: reasonable DuCom combination is made for small reactors and with many gas turbines.

In order to characterize the competitiveness, another function called "savings" has been defined.

$$C_{savings} = Min(C_{NUC}, C_{GAS}) - C_{DuCom} \qquad (7)$$

At points where $C_{savings}$ are positive, the *DuCom* option is the most competitive. Otherwise, one of the two energy sources (nuclear or Gas) will be the most competitive. Points with $C_{savings}$ equal to zero define the *indifference surface*.

Figure 7 shows the indifference surface in the same ranges that figure 6. Different points of view of same surface are presented in figure 8 for better appreciation of its 3-D aspects.

The indifference surface can be interpreted as two individual surfaces. The upper surface belongs to $C_{savings} = C_{NUC}$ and the lower surface belonging to $C_{savings}=C_{GAS}$. The regions delimited by these two surfaces $C_{savings}$ are positive where the *DuCom* option is more competitive.

Figure 9A the plotted region indicates the competitiveness region in ($\$_{GAS}$, Pot_{NUC}). For low gas prices, DuCom competes using large nuclear reactors. For high gas prices, the use of the "pure" nuclear reactor will be the cheaper option.

Figure 9B shows the competitiveness region on the (Pot_{NUC}-Pot_{GAS}) plane for **low** gas prices. *DuCom* competes using high power reactors combined with high gas turbine powers.

Figure 9C shows competitiveness region on(Pot_{NUC}-Pot_{GAS}) plane for **high** gas prices. *DuCom* is competitive when coupling small reactors with a wide range of gas turbines.

The IREP2.0 code predicts up to 13% of cost savings and 10mill$/kWh for the *DuCom* combination of 100MWth nuclear reactor power with 50MWe gas turbine power for 6$/GJ of gas price.

CONCLUSIONS AND FUTURE

The code *IREP 2.0* was presented, showing capabilities for detailed economic evaluations of hybrid gas-nuclear power plant. The new version of IREP has been based on more accurate models and extended analysis.

The combination of nuclear-gas does not impact on the original engineering parameters, saving considerable money and engineering time. This implies reusability of many reactors already constructed and vice-versa.

The separability of the plants makes possible partial generation using only the gas turbine within a year with still-in-construction nuclear reactor. From the financial point of view this very convenient.
Future analysis should quantify the increase in total load-factor by means of crossed probability analysis of availability. The total cost of electricity generation will then be more robust against seasonal gas price fluctuations and future limitation on greenhouse effect gas emissions.

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IAEA-SR-218/33

THE ECONOMICS OF MEDIUM-SIZED REACTORS IN DEREGULATED MARKETS

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Abstract

The prospects for the deployment of small and medium reactors in deregulated markets is crucially dependent on demonstrating that such reactors are economically competitive with alternative power sources. Other factors, such as safety, reliability, waste arisings, environmental impact etc are also important to get right, but unless the economics are competitive, private investment is unlikely to be available for any reactor system. This paper examines the economics of the Westinghouse AP-600, which is the only medium-sized evolutionary PWR in the world today which has achieved regulatory approval. The paper presents an economic analysis of an AP-600 which shows that medium-sized systems can be economic in a competitive market and identifies the economic barriers that need to be overcome for their deployment to become a reality. The paper shows how sensitive the overall generating cost can be to the perceived financial risk, which helps to highlight where attention most needs to be focused.

1. INTRODUCTION

The Westinghouse AP-600 [1] is an advanced Pressurised Water Reactor (PWR) of 600 MWe output based on proven technology combined with many passive engineering safety features that is designed to provide:

1. Simplified design and construction compared with the current generation of PWRs, with lower capital investment costs and a shorter build time.
2. Simplified maintenance requirements giving a reduction in operating and maintenance (O&M) costs.
3. Increased operating margins achieved through a lower core rating than current generation PWRs.
4. Smaller unit size to better match utility requirements.

AP-600 has already received design certification from the US Nuclear Regulatory Commission (NRC) and could in principle be constructed immediately in the US without further regulatory process. The AP-600 is the only medium or small plant ready for construction that has achieved such regulatory acceptance.

An analysis of the total generating cost of AP-600 in the UK's deregulated market has recently been published [2] which shows that AP-600 is close to being competitive compared with combined-cycle gas turbine CCGT generation, which in the UK defines the lowest generating cost baseline. The analysis in [2] was actually based on a four AP-600 unit power plant generating 2520 MWe. This assumption was made to allow comparison with earlier studies for a previous generation plant with twin 1310 MWe plants generating a total of 2620 MWe. This paper presents an adaptation of the work in [2] and uses it to illustrate some of the issues that arise in considering the introduction of a medium-sized reactor into a deregulated market.

2. AP-600 GENERATION COST

A recent study [2] has analysed the competitiveness of the AP-600 in the UK's deregulated market. The study estimated the total generating cost for a power station consisting of four AP-600 units generating a total of 2520 MWe. For conventional and evolutionary PWRs in the UK, there is a clear economic advantage for a total capacity exceeding 1 GWe because of minimum staffing requirements and other fixed cost items that give savings for large capacities. AP-600 could meet the requirement for > 1 GWe capacity as a twin unit plant, or in the case considered, a four unit plant. A four unit plant was considered in [2] largely to facilitate comparison with an earlier economic assessment of a twin unit power plant based on a replica of the Sizewell B PWR design with 2620 MWe total capacity[3]. Table I shows the input assumptions to the economic assessment. The original data in [2] were presented in UK Sterling (£) and these have been converted to US Dollars assuming an exchange rate of 1.5 $/£.

TABLE I. ECONOMIC ASSUMPTIONS FOR A 4-UNIT AP-600

Item	Value	Item	Value
Construction time	6y	Total fuel loading	268 tonne U (4 × 600 MW units)
Lifetime	40y	Fuel cycle length	24
Capital cost	$3165m	Fraction of core discharged per reload	33%
Annual operating cost	$141m	Decommissioning cost	$548m
Fuel purchase cost	$1.88m/tonne U	Delay between shutdown and decommissioning	40y
Fuel disposal cost	$1.2m/tonne U	Electrical output	2520 MW
Delay between fuel discharge and disposal	40y	Load factor	90%

The capital cost and operating and maintenance (O&M) cost in Table I are based on detailed engineering cost estimates for an n'th-of-a-kind plant sited in the US, with an allocation in the capital cost estimate for first-of-a-kind costs assumed to be shared over 10 AP-600 units. The capital and operating costs can be considered applicable in any country whose licensing requirements are based on those of the USNRC, although site specific and country specific adjustments would need to be taken into account. The capital and O&M cost estimates have been carried out very thoroughly, with major fabricated parts having been subjected to external tenders and a detailed probabilistic analysis of uncertainty bounds carried out. The cost estimate included all the support facilities needed to operate the plant. The results were generated in two Phases. Phase I involved the production of a very detailed cost estimate containing approximately 6000 line items. Phase II consisted of evaluating the uncertainty of the cost estimate using probabilistic cost ranges for the input parameters. The AP-600 capital and O&M costs are therefore considered very robust and defensible. The fuel purchase cost, fuel disposal cost and decommissioning cost are all considered to be pessimistic figures that are intended to allow for adverse future trends, such as an increase in long term uranium ore prices. The 90% load factor is similarly considered pessimistic, since the engineering assessments indicate that a higher value should be achievable.

The data in Table I were input to an economic model and used to calculate the levelised cost of generation as a function of the investment discount rate. The levelised cost is calculated by dividing the present value of all costs incurred in the lifetime of the plant by the discounted electricity generation over the lifetime of the plant, both to some common reference date. Table II gives the levelised generating costs for a range of discount rates from 0 to 15%. Note that in [2] the range considered was from 10% to 15% and Table II has extended the range down to 0%. The units are $/MWh, which are equivalent to the more commonly used unit of mills ($\$10^{-3}$) per kWh.

The analysis summarised in Tables I and II does not actually apply to any specific site. Site specific factors can be very significant, with grid connection and transmission charges varying geographically. For example, in the UK such site specific considerations can make a difference of more than 5 $/MWh to the total generating cost.

TABLE II. TOTAL GENERATION COST FOR 4-UNIT AP-600

Discount rate (%)	Levelised total generating cost ($/MWh)
0	16.8
5	23.8
8	30.0
10	34.9
11	37.6
12	40.4
13	43.4
14	46.5
15	49.8

The levelised generating cost is very sensitive to the discount rate. By way of comparison, in the UK market the lowest price generation has been set by CCGT plants. In late 1999, gas prices were at an all-time low and the total cost of generation from CCGT was approximately 30 $/MWh. Table II shows that an n'th of a kind multiple unit AP-600 plant could be economically competitive compared with CCGT at discount rates up to 8%, with low gas prices and without any subsidies such as a carbon tax assumed. This is a considerable improvement over previous generation PWR designs and results from the design and operational simplifications noted earlier. However at the higher discount rates likely to be required by private investors in deregulated markets, a multiple unit AP-600 plant must include levelising influences such as a carbon emissions levy to be competitive with CCGT in a scenario with low gas prices. Since 1999 however, world gas prices have risen significantly and the future cost of generation from CCGT in the UK is likely to be significantly higher than the 30 $/MWh low. In a market with volatile gas prices, most likely trending upwards with time, the gap between AP-600 and CCGT at commercially relevant discount rates is likely to narrow.

It is informative to examine the sensitivity to discount rate in more detail, because it leads to a clearer understanding of where attention should be focused. This is the subject of Sections 3 and 4:

3. ORIGIN OF THE SENSITIVITY TO DISCOUNT RATE

To see where the sensitivity to discount rate arises, it is helpful to break down the overall generating cost into its four principal components: capital, O&M, fuel and decommissioning, as this shows which are most sensitive to financial risk (note that it is the convention to allocate all costs incurred before first power production to capital, and all costs incurred after last power production to decommissioning). Table III gives a breakdown of the overall generating cost at 10% discount rate from Table II into these four components:

TABLE III. BREAKDOWN OF AP-600 TOTAL GENERATION COST FOR 10% DISCOUNT RATE CASE

Component	Component cost ($/MWh)
Capital	21.0
Operating and maintenance (O&M)	7.1
Fuel (including fuel disposal)	6.6
Decommissioning provision	0.2
Total	**34.9**

The capital cost component accounts for 60% of the total generating cost and is also the most sensitive to financial risk as can be seen from its mathematical formulation:

The levelised capital cost L_C in units of $/MWh is obtained by dividing the discounted cost of building the plant C_{con} ($m) over the construction period t_C years by the levelised income from electricity sales over the economic lifetime of the plant t_E years:

$$L_C = (C_{con}/Q)\, D_C(r, t_C, t_E) \text{ \$/MWh} \qquad \textbf{Eq 1}$$

where Q is the (undiscounted) annual electrical output in MWh, r is the fractional discount rate and $D_C(r, t_C, t_E)$ is the discount factor for the construction levelised cost. D_C takes two forms depending on whether the capital cost and electrical output are discounted discretely (ie as a block once a year) or continuously:

Discrete discounting form:

$$D_C(r, t_C, t_E) = (1/t_C)(1+r)^{t_C} [1 - 1/(1+r)^{t_C}]/[1-1/(1+r)^{t_E}]$$

Continuous discounting form:

$$D_C(r, t_C, t_E) = (1/t_C)\exp(rt_C) [1 - \exp(-rt_C)]/[1- \exp(-rt_E)]$$

The factor $(1/t_C)$ simply converts the total capital cost of the plant to the annual expenditure during the construction phase (we are simplistically assuming a uniform spend profile during the construction period). The factor $(1+r)^{t_C}$ allows for the discount effect of the delay of t_C years between start of construction and first power production, while the remaining factors give the present value of the construction costs and electricity production, which are incurred over t_C and t_E years respectively.

Figs 1 to 4 show how D_C varies with t_C and t_E for discount rates of 0, 5, 10 and 15% respectively. To see how to apply these figures consider for example the value of the discount factor from Fig 3 corresponding to t_C = 6 years and t_E = 40 years, which is 0.1315. Multiplying D_C by the capital cost $3170m and dividing by the annual electrical output of the plant $Q = 0.9 \times 2520 \times 365.24 \times 24$ MWh $= 1.988 \times 10^7$ MWh gives the levelised capital cost of 21.0 $/MWh as in Table III.

At zero discount rate, as shown in Fig 1, there is no dependence on the construction time t_C and the levelised cost varies as $1/t_E$. This is understandable, since at zero discount rate future electricity generation counts as much as electricity generated today. For non-zero discount rates, the construction time becomes increasingly more important, with the levelised cost increasing for longer construction times. The curves flatten out at higher economic lifetimes, especially at high discount rates, because in this case later electricity generation doesn't contribute much to the total because of the effect of discounting.

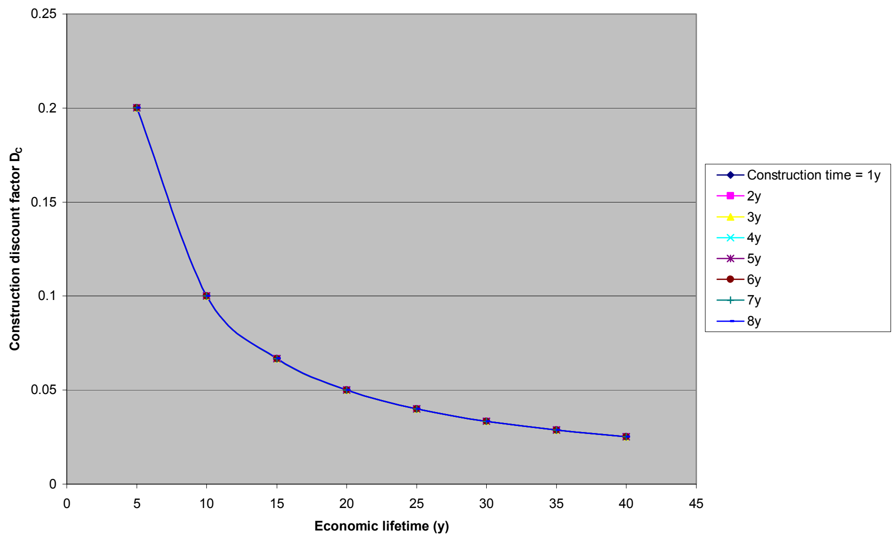

FIG. 1: Discount factor for capital versus t_C and t_E – 0% discount rate.

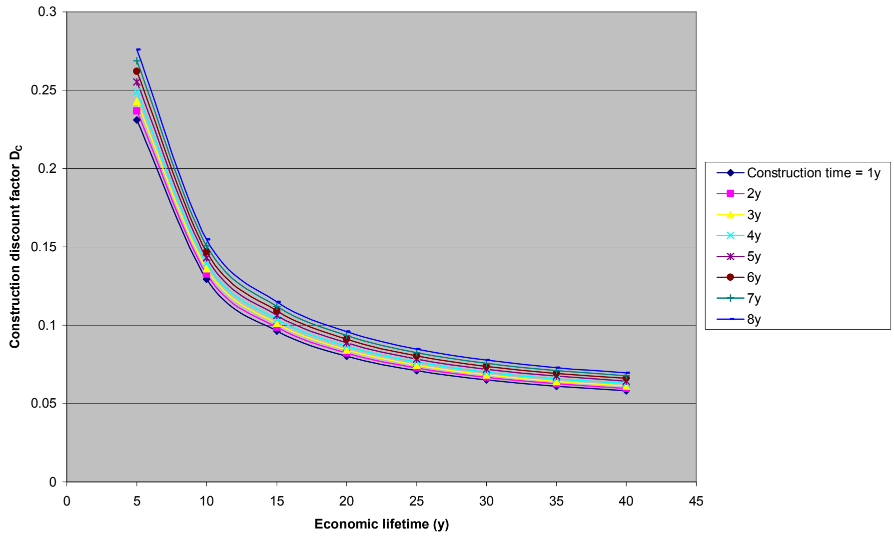

FIG. 2: Discount factor for capital versus t_C and t_E – 5% discount rate.

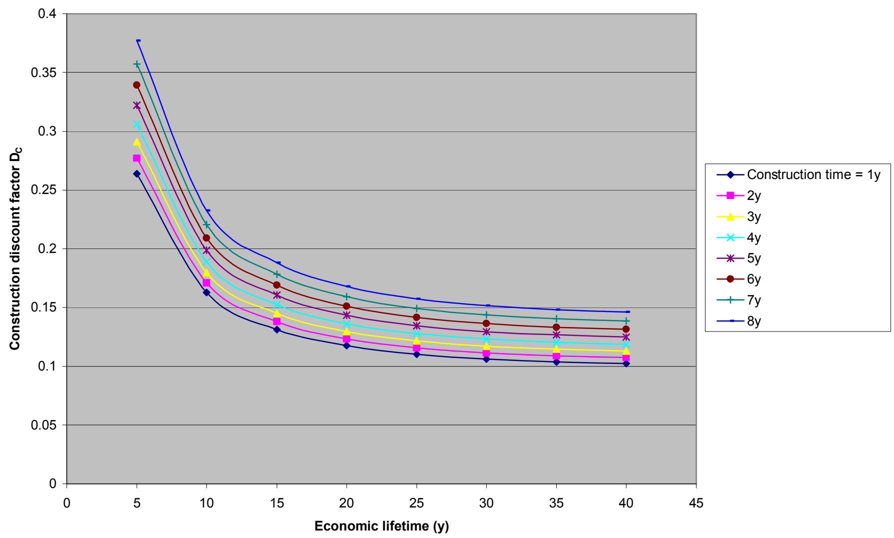

FIG. 3: Discount factor for capital versus t_C and t_E – 10% discount rate.

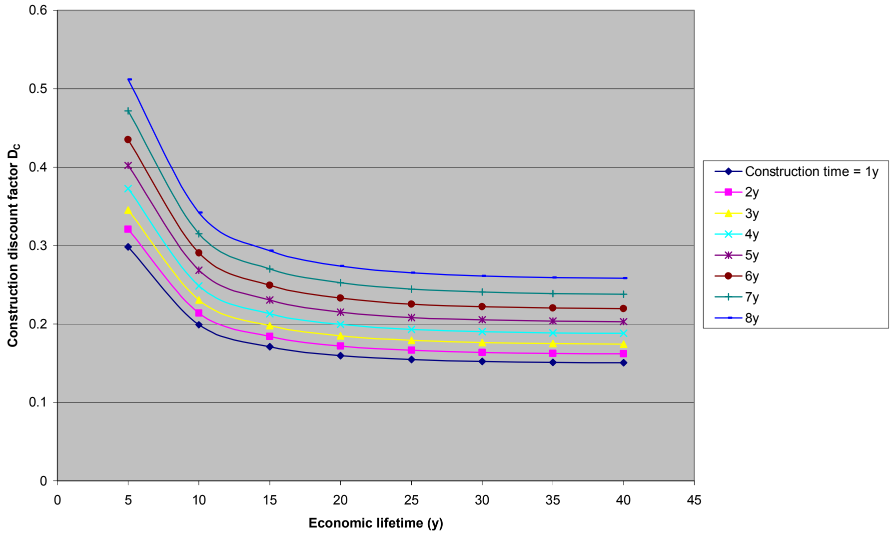

FIG. 4: Discount factor for capital versus t_C and t_E – 15% discount rate.

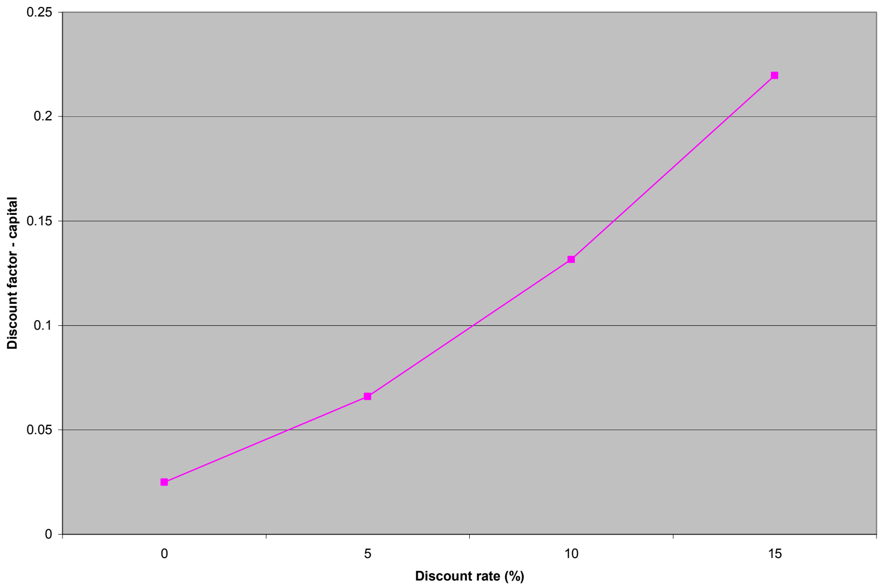

Fig 5: Sensitivity of levelised capital cost discount factor to discount rate.

Fig 5 illustrates the sensitivity of the capital discount factor to discount rate for the case of a 6 year construction time and a 40 year economic lifetime, consistent with the AP-600 base assumptions. The effect of increasing the discount rate from zero to 15% is to increase the capital discount factor by almost 900%! This is the principal cause of the sensitivity of overall generating cost to discount rate in Table II.

The levelised O&M cost is insensitive to the discount rate. The reason is that the time profile of O&M costs is identical to the time profile of electricity production (assuming that both the annual O&M cost and the annual electrical output are constant) and the two discount factors cancel.

The fuel cost shows some slight sensitivity to discount rate because of the fact that uranium ore, conversion, enrichment and fabrication services need to be paid for in advance, while electricity production is spread over the irradiation dwell time of the fuel. Bearing in mind that fuel costs only represent around 20% of the total generating cost (Table III), the impact on total generating cost is minimal.

The decommissioning levelised cost similarly contributes almost nothing to the sensitivity of overall generating cost to discount factor. A favoured approach to decommissioning is to calculate a decommissioning provision at a low discount rate and the accounting assumption is that the provisioning rate does not vary with the commercial discount rate. The provisioning approach is designed to ensure that sufficient funds will exist to cover the eventual decommissioning costs allowing for the costs to escalate. Using a low provisioning rate is appropriate because it assumes that the money will be put into safe investments that can be expected to give a low rate of return. A value of 2% is assumed to apply here. Since the provisioning rate is fixed, there is no direct sensitivity of the decommissioning levelised cost to discount rate. In any case, examination of Table III shows the decommissioning provision to be very small, so that any sensitivity would not affect the overall generating cost significantly.

From this discussion, we can now see that the sensitivity of overall generating cost to the discount rate is largely accounted for by the capital component. From Figs 1 to 4, it is clear that the construction time is an important sensitivity; the impact of reducing from 6 to 5 years, for example, is to reduce the levelised capital cost by 5% at 10% discount rate and by 8% at 15% discount rate. A much larger potential sensitivity is to the economic lifetime; for economic lifetimes less than 20 years, there is the potential to more than double the capital cost component. This leads to the question of financial risk and how it can penalise the economics of nuclear in deregulated markets:

4. QUANTIFYING FINANCIAL RISK

Of all the obstacles to the deployment of new nuclear generation in a deregulated market, the question of financial risk is the most significant. The problem is to secure the large investment necessary for new nuclear build when there are a large number of unhelpful factors, including:

- Long lead time for construction.
- Long time for which the reactor must operate to recover its capital cost.
- Risk of external events or political decisions affecting the ability of the plant to operate.
- Risk of plant breakdowns or under-performance limiting the income from electricity sales.
- Risk of regulatory changes affecting plant operation.

In many cases the alternatives to investment in nuclear, such as CCGT plants, are characterised by short build times, short payback periods, lack of sensitivity to external and regulatory changes and are sensitive only to long term gas prices.

It is therefore easy to see how nuclear might be seen by investors as representing a high financial risk. It is possible to raise finance for high risk projects from private investors but they naturally demand high rates of return in compensation. The impact of the perceived high investment risk of nuclear is therefore to specify a high target rate of return on investment, which equates to a high discount rate in the economic analyses. We have seen in this paper how sensitive the economics of nuclear plants are to the discount rate. The usual approach adopted in economic studies such as this one is to treat the discount factor as being imposed externally and to avoid discussion of what discount rate to apply; it is normally left to market analysts to specify the appropriate discount rate that investors might find acceptable. In this final section we will establish the quantitative relation between the discount rate and financial risk.

We need an equation to replace Eq 1 which gives the levelised capital cost for the project taking account of the probability of external events which lead either to the cancellation of the project before construction is complete or to premature shutdown during operation. This is now a stochastic problem so that operation of the reactor to the end of its economic life is not certain and the levelised capital cost L_C must be replaced by the *expectation value* of the levelised cost $< L_C>$, this being the average value of the levelised cost for a large number of Monte-Carlo trials.

The mathematics is identical to the survival of neutrons in an absorbing medium, the removal of neutrons being analogous to non-survival of the project due to the external impacts. It is useful to define external risk constants analogous to the decay constant λ for a radioactive nuclide. We will assume the general case where a different external risk applies during the construction period to that which applies during the operating period. We will define these as ρ_1 and ρ_2 respectively, both measured in units year^{-1}. The probability of the project surviving the construction period is $\exp(-\rho_1 t_C)$ and then the probability of subsequently surviving the full operating period is $\exp(-\rho_1 t_C) \times \exp(-\rho_2 t_E)$. Eq 1 is then replaced by:

$$<L_C> = (C_{con}/Q) <D_C>(r, t_C, t_E)\$/\mathrm{MWh} \qquad \textbf{Eq 2}$$

where, with the continuous form of the discount equations

$$<D_C>(r, t_C, t_E) = (1/t_C)\exp(r+\rho_1)t_C \times [(r+\rho_2)/(r+\rho_1)\ ] \times$$

$$[1-\exp-(r+\rho_1)t_C]/[1- \exp-(r+\rho_2)t_E]$$

In Eq 2, the correct discount rate to use is the *risk-free* value, since we are attempting to allow for the impact of risk explicitly. Note that in the case of zero external risk ($\rho_1 = \rho_2 = 0$), we recover the continuous discounting form of Eq 1. Fig 6 plots the ratio [$<L_C>$from Eq 2]/[L_C from Eq 1] versus external risk, assuming that the external risk constant that applies during the construction period, ρ_1, is identical to that which applies during the operating period ρ_2 and we will denote both by ρ. The ratio $<L_C>/L_C$ is the ratio of levelised capital cost for a certain level of external risk relative to the *risk-free* levelised capital cost. For the example of 6 years' construction, 40 years' operating appropriate to AP-600 and assuming a risk-free discount rate of 5%. Rather than plot the external risk constant ρ on the horizontal axis, which is not immediately meaningful, the horizontal axis in Fig 6 gives the probability that the project survives to the end (ie to time (t_C + t_E) years = $[1 - \exp-\rho(t_C+t_E)]$), which is a more useful measure.

Fig 6 shows $<L_C>/L_C$ increasing as external risk increases. $<L_C>/L_C$ from Eq 1 is approximately 1.27 at the point at which there is judged to be a 50% risk of premature termination of the project due to external events during the 46 year lifetime. This corresponds to a total generating cost penalty of 5.27 \$/MWh. Such events could constitute premature failure of a major component leading to closure, generic failure in a sister plant, failure in an unrelated plant (ie. such as the Chernobyl explosion leading to the closure of some Western plants), political policy changes leading to premature closure (as in Bärseback) etc.

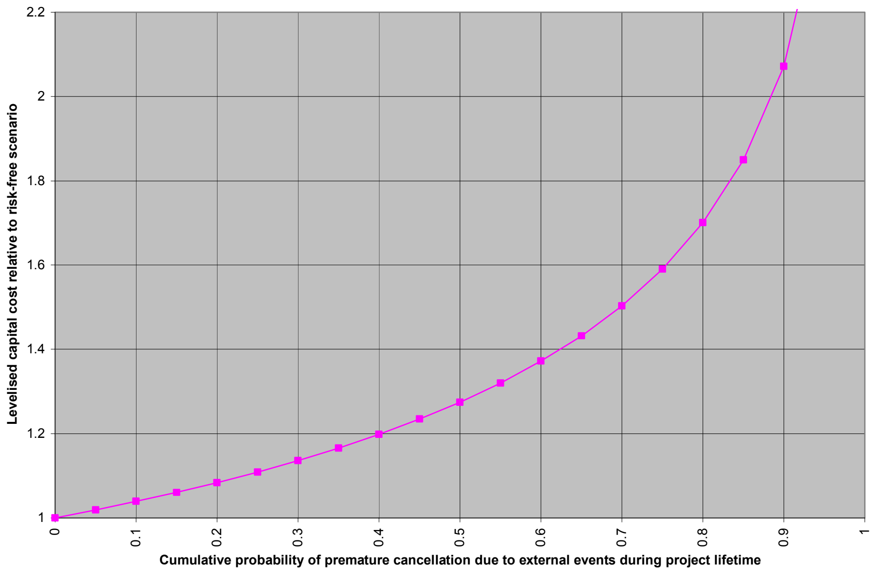

FIG 6: Ratio of expected levelised capital cost/risk-free levelised capital cost as a function of external risk – 6 year build time, 40 year economic lifetime, 5% risk-free discount rate.

Fig 6 is only intended to illustrate the sensitivity. It is difficult to know exactly what the risk-free discount rate should be. To compare with non-nuclear alternatives, it might be most appropriate to choose the risk-free discount factor to specifically exclude only those risks which are specific to a nuclear project. It is even more difficult to judge how much external risk is appropriate. Nevertheless, the message is clear from Fig 6 that the risk element has the potential to increase the expected value of the discount rate by a substantial amount and this is penalising to the economic competitiveness of nuclear. The sensitivity is potentially much greater than that due to varying construction times.

What is there to be learned from this approach. There are four key points:

1. It is very important that the plant design should as much as possible exclude the possibility of major plant failures leading to premature closure.
2. It is important that the nuclear industry does as much as possible to exclude catastrophic failures in *other* plants to avoid the political repercussions of another Chernobyl.
3. It is important to work on ensuring that the *perception* of investors as to the external risks to nuclear is in accordance with the technical reality.
4. It is crucially important to work to ensure that there is long term political stability in the deregulated market, preferably such that successive governments are committed to ensuring long term stability.

In successfully addressing these points the nuclear industry would have removed the biggest economic sensitivity is removed from the economics analysis and the biggest single barrier to new reactor build in deregulated markets.

CONCLUSIONS

This paper has demonstrated how an evolutionary medium size PWR, the Westinghouse AP600, is close to being competitive in the deregulated market, compared with the lowest cost generation set by CCGT plants. Relatively modest adjustments, such as carbon tax levelisation scheme, higher gas prices or plant uprating could be sufficient to make AP-600 fully competitive economically.

In markets such as the UK, where there is a strong grid infrastructure, it is clear that medium-sized plants will only be viable when multiple units are installed. Siting considerations are also important, with incentives/disincentives that can amount to as much as 5 \$/MWh applying depending on where the plant is sited relative to where the power is required.

We have seen that the capital component represents the biggest single contribution to the levelised generating cost (typically > 60%). The capital cost is also the most sensitive to the economic discount rate assumed and gives rise to the bulk of the sensitivity of overall generating cost to discount rate. The connection between the unique risk factors of nuclear generation and the discount rate used by private investors to judge a project has been emphasised and the sensitivity to project risk has been identified. It is crucial for nuclear projects to ensure there is long term political stability that will allow them to proceed with minimal risk of premature closure; any such risk translates directly into a levelised generating cost penalty that makes raising the private investment needed for new nuclear build more difficult. Thus while technical fixes can help ensure competitiveness, it is even more important to work to ensure the correct political environment exists to allow future nuclear build to thrive in deregulated markets.

ACKNOWEDGEMENTS

The author wishes to acknowledge Gary Mangham of BNFL Research & Technology for independently deriving Eq 2 and verifying it with a Monte-Carlo simulation.

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ECONOMIC COMPETITIVENESS OF SMRs IN BELARUS

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Abstract

The results of the different feasibility studies have shown that an optimum way out of the energy crisis in Belarus in the short-term is the improvement of end-use energy efficiency and increasing the utilization of the domestic fuel resources. Nuclear power option is also attractive from the economical point of view in the long-term. Taking into account that heat generation accounts for about 50% of the total Belarus energy balance, a comparative analysis of economical parameters of SMRs for electricity and heat generation has been carried out. For this feasibility study the BALANCE module of the computer code ENPEP was used. Two scenarios for the optimal expansion plan up to 2020 were considered. The results of the calculation show that the implementation of nuclear power units would allow decreasing electricity and heat generation cost if the capital cost of the nuclear units is approximately 1300 USD/kW.

1. INTRODUCTION

The Republic of Belarus, a former member of the Soviet Union, has become a powerful energy-intensive industrial country. From 1960 to 1990 electricity consumption increased by a factor of 20 and the heat supplied by district heating network was increased by a factor of 17. Naturally it required the development of appropriate energy base. Not possessing domestic fuel and power resources, Belarus was oriented to nuclear power. But the first nuclear power plants (NPP) were built close to, but outside its borders (Ignalina in Lithuania, Smolensk in Russia, Chernobyl and Rovno in the Ukraine). The total capacity of these NPPs constituted about 10000 MW that exceeded the capacity of all Belarus power plants on fossil fuel by more than a factor of 1.5. In 1983 the construction of nuclear CHPP was begun not far from Minsk (the first generation of 2000 MW), then the construction of Belarus NPP was planned with the capacity of 6000 MW in Vitebsk region. The accident at the Chernobyl NPP stopped the Nuclear Power Program in Belarus.

After having declared the independence the Republic of Belarus was forced to import 90% of fuel consumed and 25% of electric energy. The deficit of peak electric capacity reached 40%. In the last few years the drop in industrial production has led to a reduction in energy consumption in the Republic of Belarus and fuel imports have covered the needs of the whole power system; about 30% of energy consumed was imported in 1999. The results of the different feasibility studies has shown that an optimum way out of the energy crisis in the short-term is the improvement of end-use energy efficiency and increasing utilization of domestic fuel resources. The nuclear power option is also attractive from the economical point of view in the long-term.

2. METHODOLOGY OF THE STUDY

The efficiency of including in the electricity generation system nuclear power plants based on large evolutionary reactors was considered in [1]. Taking into account that heat generation makes up about 50% of the total Belarus energy balance, a comparative analysis of the economical parameters of small and medium size reactors for electricity and heat generation has been carried out. For this feasibility study the BALANCE module of the computer code ENPEP was used.

The principal objective of the BALANCE Module is to project the supply/demand balance for up to 75 years. BALANCE facilitates the designing of a comprehensive model of the energy-producing sector and energy - utilizing activities in the energy-consuming sectors of the country. The BALANCE Module processes a representative network of all energy production, conversion, distribution, and utilization activities in a country or region, as well as the flows of energy and fuels among these activities.

For this study a simplified network has been developed. The Electricity Generation Node of the network includes all units of Belarus electricity generation system and co-generation power plants for comparison of the heat supply options. The BALANCE Model does not allow the modeling of co-generation technology directly. Nevertheless, there are different ways for modeling combined generation electricity and heat. One of them is the adaptation Multiple-Output-Link Node for the analysis of this type of technology. In this case the input link is a fuel energy flow and output links are electricity and heat (co-generation node).

3. DESCRIPTION OF SCENARIOS AND INPUT DATA

Two scenarios for the optimal expansion plan up to 2020 were considered. For all scenarios the following technologies for electricity and heat generating were included:

- Conventional co-generation technologies (natural gas and oil);
- Boilers for heat generation (natural gas and oil);
- Medium sized reactors with liquid metal as a coolant for electricity and heat generation (MSLM).
- Medium sized reactors with water as a coolant for electricity and heat generation (MSW).
- Condensing turbines with natural gas as fuel.

The parameters of Russian design medium sized reactors were the following [2,3]:

	Water reactor	*LM reactor*
	235	585
Thermal capacity, MW	63	110
Heat	50	250
O&M cost, cents/kWh	0.34	0.54
Load factor	0.8	0.8
Capital cost, USD/kW		
Scenario 1	1300	1300
Scenario 2	1800	1800
Life time, year	40	50

The Belarus electricity generating system depends largely on the import of natural gas and crude oil. Presently and in the near term future there are no other options to import energy resources from Russia to Belarus. The natural gas and crude oil price forecast for the Belarus market was based on the assumption that in 2005-2010 Belarus utilities will have to purchase energy resources at the average European price, for the following reasons:

- Exhaustion of existing oil and natural gas fields
- Unavailability of investments needed to access more economically attractive fields.

This will force the Russian suppliers of natural gas to increase the relatively low price inside Russia and the price of natural gas to be exported to Belarus. In accordance with "Concept of Energy Sector Development in Russian Federation" the natural gas price for Russian consumers has to be increased up to 1.7 times to 2005. Undoubtedly, it will lead to increasing natural gas price in Belarus and the price will approach the European price. The forecast of average European price of natural gas and fuel oil was calculated using information [4]. Table 1 contains the results of this calculation.

TABLE I. FUEL PRICE FORECAST

Year	Natural gas		Fuel oil		Nuclear
	USD/1000 m^3.	USD/Gcal	USD/ton	USD/Gcal	USD/Gcal
2000	79.6	9.82	95.3	9.82	2.35
2005	110.0	13.57	132.0	13.60	2.41
2010	123.3	15.22	148.0	15.25	2.47
2020	150.0	18.51	180.0	18.55	2.60

4. RESULTS AND ANALYSIS

The BALANCE module allows modeling of the selection of energy from alternative sources of supply according to price or other preference. To model the decision process, the network includes nodes which have two or more input links and one output links. Decision nodes select the amount of energy to be supplied from alternative sources (the input links of the nodes) at various points of the network, and route the energy to satisfy energy flow requirements of the output links of the node. Price and quantity equations are associated with a decision node. The quantity equations equates the total energy flow on the output links of the node the total energy flow on the input links to the node; energy flow is conserved at a decision node. The price equation relates the price of the energy flow on the input links of the node to the price of energy flow on the output links of the node. In addition, several other equations indicate the share of heat selected from the input links to the nodes. Shares are based on the relative price of heat from alternative sources.

Decision nodes are positioned in the network to indicate the point at which heat choices are made from alternative supply sources. The share of energy in decision node is in general a function of the relative price on the links. A higher price on an input link results in smaller share of quantity allocated to input link. This algorithm allows the modeling of the energy market for end-use consumers.

Fig. 1 shows the results of a calculation of the trade-off between different technologies for electricity generation. The result of the BALANCE run for Scenario 1 shows that electricity to be generated by nuclear power units is more preferable for end-use consumers. In the end of period of calculation the market share of electricity to be generated by MSW is higher than the market share of electricity to be generated by MSLM. It means that a medium size reactor with water as coolant is more economically attractive.

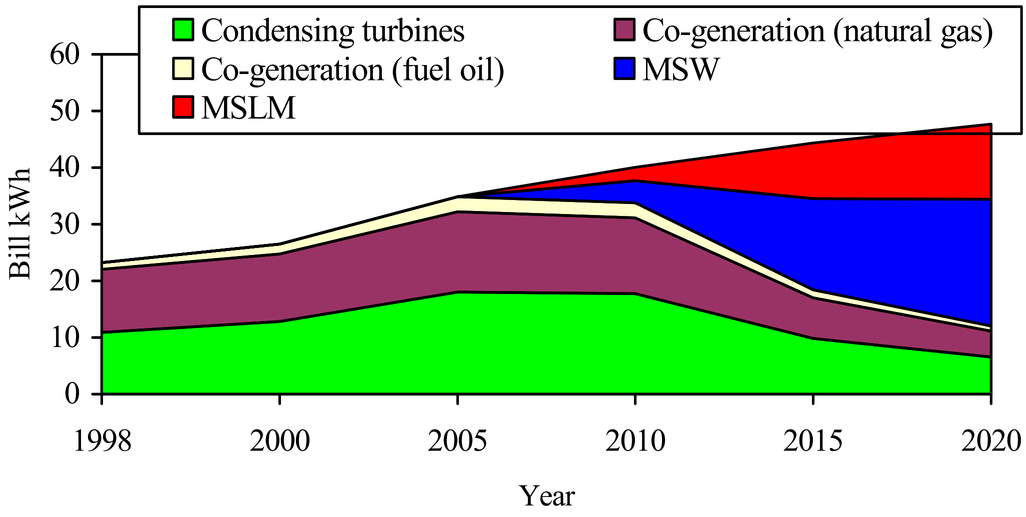

FIG. 1. Expected Electricity Generation by Technology Type.

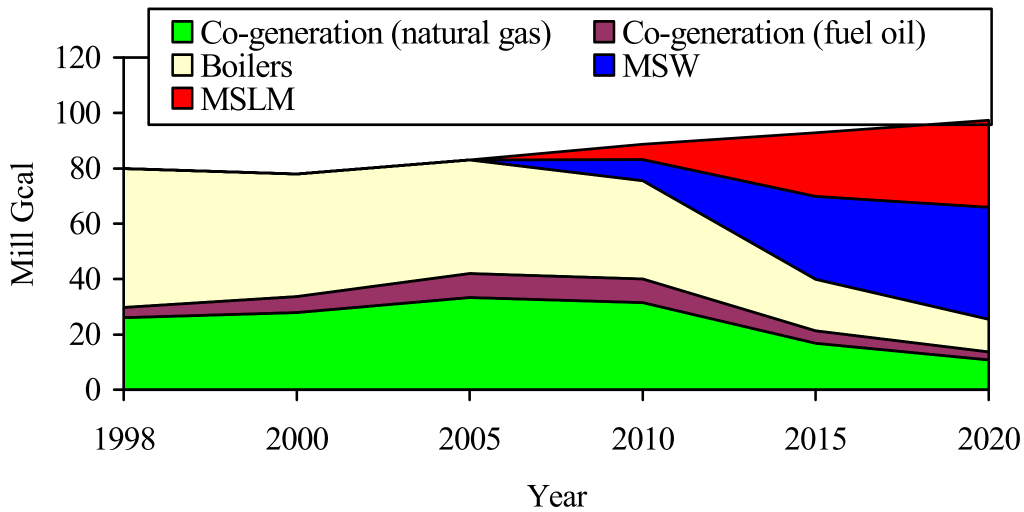

FIG. 2. Expected Heat generation by Technology Type.

Fig. 2 shows the results of a BALANCE run for the same Scenario. The calculation shows that heat generated by nuclear power reactors is more competitive in the heat market compared with conventional boilers. The market share calculation for Scenario 2 shows that electricity and heat produced by nuclear power reactors are not competitive in the energy market for end-use consumers.

It also can be illustrated by comparison of electricity and heat prices for end-use consumers, which have been calculated with taking into account the market share of different technologies for electricity and heat generation. The results of this calculation for both Scenarios are shown in Fig. 3 and 4. In comparison with non-nuclear option, implementation of the nuclear technologies for electricity and heat generation allows a decrease of electricity and heat prices up to 5 % and 6 %, respectively. If the capital cost of nuclear units equates to approximately 1800 USD/kW, the nuclear option is not economically attractive for electricity and heat generation. However, more detailed WASP analysis of the utilization of nuclear technologies for electricity generation indicates that the nuclear option can be economically competitive even if the capital cost of the nuclear units marginally exceeds 1800 USD/kW [1].

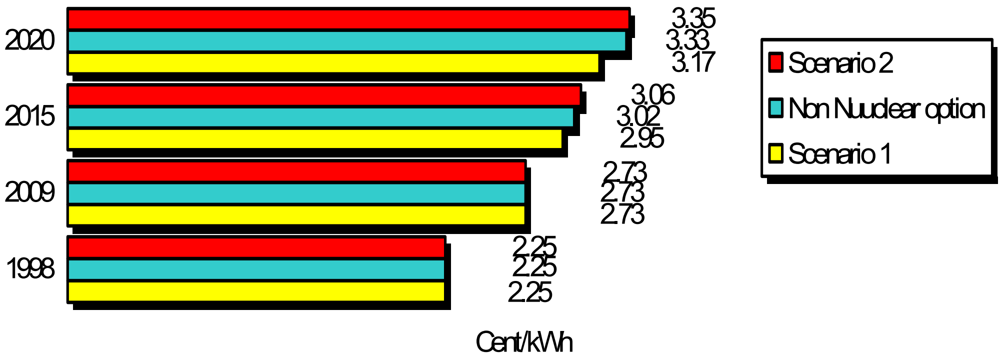

FIG. 3. Expected Electricity Price.

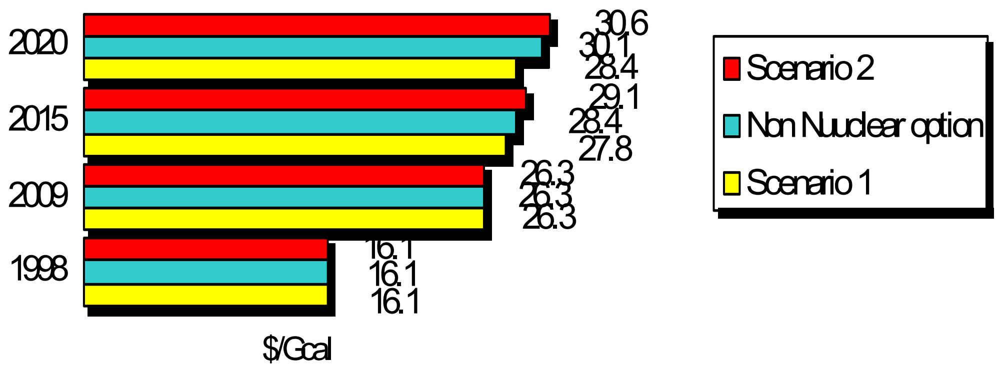

FIG. 4. Expected Heat Price.

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SMALL AND MEDIUM REACTOR FUEL CYCLES

(Session 4)

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AN INNOVATIVE FUEL CYCLE CONCEPT WITH NON-PROLIFERATION AND WASTE CONSIDERATIONS FOR SMALL AND MEDIUM SIZED REACTORS

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Abstract

Small and medium (S&M) sized reactors are attractive for countries or utilities, which have small electricity grid systems and needs to provide both electricity and process heat to remote communities. S&M reactors must be cost competitive, and their operation must adhere to the international safety standards. In addition, new approaches, which will strengthen the nuclear fuel cycle requirements in areas of waste and non-proliferation, are needed to help gain public acceptance and support.

1. INTRODUCTION

The global demand for energy will increase substantially over the next fifty years. To provide for such a demand and at the same time to limit carbon and other greenhouse gas emissions and to conserve the valuable natural resources, nuclear energy would have to be a major contributor to the energy supply. Furthermore, the demand will come from one third of the world population, which currently does not have access to electricity, and from communities in rural and remote areas. Considering the infrastructure requirement alone, small and medium (S&M) sized nuclear systems would be preferable to meet this specific demand.

Nuclear power currently makes up about 16% of the global electricity consumption. Most of the nuclear capacities are now provided by large nuclear power plants (>600 MWe). Most of these large reactors are providing base-load electricity to their customers and some of them have been amortized of their capital expenditure, and hence can be operated favourably in a competitive electricity generating market against other energy sources, such as natural gas. Despite the incident at Three-Mile Island in 1979 and the catastrophic accident in Chernobyl in 1986, the nuclear industry and utilities have been operating reactors for several decades, acquiring many reactor-years of safe operation experience.

Since the dawn of the nuclear era, there has been a concern for misuse of nuclear materials intended for peaceful purposes by its owners, and for theft or diversion of the materials by rogue nations or terrorist groups. The establishment of the International Atomic Energy Agency (IAEA) in 1957 was intended to prevent the potential spread of nuclear-weapons materials and capability. The Non-Proliferation Treaty (NPT) of 1968, signed now by more than 170 countries also served the purpose of limiting the number of weapons states to those already-declared before the initiation of NPT. Nevertheless, the countries' desire to acquire the fuel cycle capabilities (technologies and facilities) for the purpose of energy independence remains a challenge for NPT to keep the regime intact and for IAEA to administer cost-effective safeguards.

The back-end of the nuclear fuel cycle begins with spent fuel discharge from the nuclear reactors. Technically, 95% of the entire radioactivity resulting from nuclear electricity generation ends up in the spent fuel itself. Whether reprocessed or not, the volumes of spent fuel and the reprocessed radioactive wastes are small by any modern industrial waste standards, and there are demonstrated methods (dry and wet) for interim storage of spent fuel and radioactive wastes. Ultimately, the spent fuel and radioactive wastes should be disposed of. Several countries have embarked on their respective national repository programs to dispose of these materials in geologic repositories. Here, the challenge is how to engage the public and stakeholders to gain their acceptance of a waste management program including a geologic repository built by a national effort, or by a co-operative regional[1] or international arrangement.

2. CHALLENGES TO SMALL & MEDIUM SIZED REACTOR OWNERS

Of the four considerations associated with nuclear power generation: economics, safety, waste and non-proliferation, the last two are fuel-cycle related and have become the most intractable techno-institutional challenge. They are the focus of this study.

2.1. Spent fuel and radioactive-waste consideration

Currently, nuclear utilities are accumulating spent fuel and/or radioactive wastes on-site or at away-from-reactor storage[2] pending on the final disposal in geologic repositories. Some utilities (and countries) with small nuclear power programs, and therefore relatively small amounts of spent fuel and radioactive wastes may have limited resources and potentials to develop their own back-end fuel cycle systems. They are facing a possibility of prematurely terminating their existing nuclear power programs.

Ultimately, spent fuel and radioactive waste would be disposed of in suitable geologic repositories. Several countries have begun their national repository programs at specified or demonstration sites, e.g. the Yucca Mountain site in the US, STRIPA in Sweden, Mol in Belgium, and Gorleben in Germany, etc. The repository technology pursued by each country is site-specific, and the timing for a repository is country-dependent. The challenge for repository development is institutional and political, e.g. how to obtain the public and stakeholders' acceptance and support for the location of a repository.

For countries with small nuclear power programs and small amount of spent fuel and radioactive wastes, and those with dense population and small geographic areas, finding a suitable site for a repository may be difficult. These countries may also have limited potentials and resources to develop their own systems for managing the back-end of the nuclear fuel cycles. Furthermore, it may not be in the interest of the international community that spent-fuel repositories are spread out all over the world, which may constitute a long-term proliferation risk.

Hence, the challenge on repository or permanent storage of spent fuel and radioactive wastes remains. It is summarized as follows:

- Location, Location, and Location NIMBY (Not In My Back Yard) concern,
- Institutional Concerns (Political, Legal, Ethical, etc.) NIMTO (Not In My Term of Office), and
- Long-term proliferation concerns and safeguards requirements for spent fuel in repository or in permanent storage.

2.2. Non-proliferation consideration

Fissionable nuclear materials are used and simultaneously generated in nuclear reactors. To help address non-proliferation concern and to provide safeguards of fissionable materials, institutional frameworks (e.g. the IAEA, the Non-Proliferation Treaty, and the Nuclear Suppliers' Group[3], etc.) were established to help facilitate the civilian nuclear programs. At the same time, key elements of the civilian nuclear fuel cycle intended for peaceful use purposes were scrutinized by regional or international safeguards regimes to prevent the potential diversion of fissionable materials for other uses.

To minimize the risk of misusing civil technologies and civil fissionable materials for non-peaceful purposes a comprehensive non-proliferation system had been developed. The system includes the following:

- **International Institutional Measures**
  (e.g. NPT, Comprehensive Safeguards Agreements, Additional Protocol, IAEA recommendation on physical protection, etc.),
- **International Verification Measures**
  (e.g. IAEA Safeguards, Regional and Bilateral Safeguards Agreements),
- **Export Controls** on nuclear materials, specific facilities and equipment, including dual-use technologies and materials (Nuclear Suppliers' Group, Zangger Committee[3], etc.),
- **National Physical Protection Measures and Materials Accounting and Control Measures.**

By and large, this existing system has shown its effectiveness. For the time being no single signatory of the NPT has tried to misuse their civil technologies and civil fissionable materials for non-peaceful purposes.

Nevertheless, significant safeguards inspection effort and traditional measures have been and will continuously be spent by regional or international safeguards and verification regime. For example, Table 1 shows the number of person-days of inspection performed by IAEA annually for typical declared facilities[4]. Since international safeguards is needed for as long as the nuclear materials remain at the facility sites regardless whether the facilities are in operation or shutdown, there will be a continuous effort and traditional measures imposed on the owners of these facilities, resulting in a continuous financial and resource commitment. This may be a burden for small utilities and facility owners operating S&M reactors.

TABLE I. SAFEGUARDS INSPECTION EFFORT, TRADITIONAL MEASURES ON DECLARED NUCLEAR FACILITIES

Type of Facility	Person Days of Inspection per Year
Light Water Reactor, no MOX	6 - 12
CANDU Reactor	45
Light Water Reactor with MOX	15 - 45
Enrichment Plant	70 - 150
MOX Fuel Fabrication Facility	~ 200
Reprocessing Plant	> 750

2.3. A need for new approaches

These two considerations (waste and non-proliferation) should be taken into account in the development and deployment of S&M sized reactors for future nuclear power generation. S&M sized reactors, by their nature are specifically developed for S&M sized nuclear power programs. Utilities or countries whose sole interest is on power generation to provide electricity and process heat and which may lack the resources to deal with the back-end fuel-cycle issues, should be relieved of the burden imposed by the “waste (spent fuel and/or radioactive wastes) and non-proliferation.”

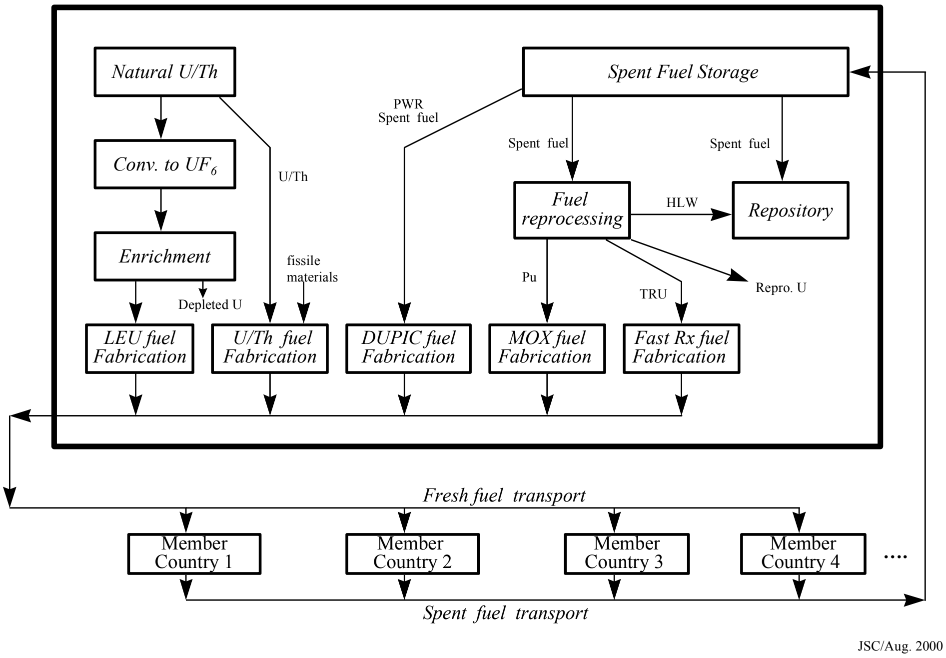
New approaches to strengthen the nuclear fuel cycle requirements in areas of waste and non-proliferation are sorely needed. The implementation of the requirements, both technical and institutional, is essential to assure a successful development and deployment of S&M sized reactors.

3. A GLOBAL NETWORK OF NUCLEAR FUEL CYCLE FACILITIES

Here, an innovative nuclear fuel-cycle concept for S&M sized reactors is presented. The concept encompasses a global network of nuclear fuel cycle facilities (Figure 1), formed by a framework of contractual agreements among companies (or countries in which the companies are operating) for the sole purpose of providing economically-competitive, safe, and proliferation-resistant fuel cycle services to nuclear reactor users, including those with S&M sized reactors.

Figure 1 A Global Network of Nuclear Fuel Cycle Facilities

This is not a physical or national boundary, it is merely formed by contractual or (framework) agreements among companies (and nations where companies are operating) to provide economically-competitive, safe, and proliferation-resistant fuel cycle services



Most of the fuel cycle facilities shown in Figure 1 are currently in operation (or under construction) in many countries. There are front-end fuel cycle facilities, including conversion, enrichment, and fabrication facilities for various fuel types, and back-end facilities, such as reprocessing and on-site spent fuel storage already available to serve the fuel cycle service needs. A few key facilities in the back-end fuel cycle, notably the regional spent fuel storage and waste repository are still absent in the global network.

The fuel cycle facilities in this global network are not necessarily owned by a country, nor need to be co-located in a so-called "fuel cycle center[5]". In fact, such a network could be formed merely by contractual agreements between two fuel cycle facilities or among a few parties. The aim is to ensure a stable and reliable supply of fresh nuclear fuel and to take the spent fuel back from reactor operators. Currently, reliable front-end fuel cycle services are provided to reactor operators in a cost-competitive manner. However, there is not yet a complete back-end fuel cycle service to deal with the spent fuel and/or radioactive wastes.

The importance of the global network of nuclear fuel cycle is to relieve the burden of waste and non-proliferation to countries/utilities operating S&M sized reactors for power generation. If fresh nuclear fuel can reliably be supplied and the spent fuel removed, the country/utility may have less incentive to pursue the acquisition of its own fuel cycle capabilities and facilities. This would be a "win-win" proposition for the reactor operators and IAEA because significant saving on safeguards inspection costs can be incurred as spent fuel assemblies are not in prolonged on-site storage to warrant the proliferation concerns.

Furthermore, if such global network can be formed, it could focus the inspection effort for international safeguards by IAEA on the fuel cycle facilities within the network. As many of these facilities are operated by and located in declared weapons states, the safeguards inspection obligations are exempt. As a result, the IAEA could achieve its goal of providing cost-effective safeguards to its Member States.

The formation of the global network depends on the success in managing the spent fuel taken back from reactor operators and the radioactive wastes generated from fuel reprocessing within the network. Many companies (and countries where companies are operating) may have political and institutional constraints which limit their ability to provide certain types of fuel cycle services, notably those in the back-end. These companies (and countries) should recognize that it is only through mutual co-operation arrangements among them (e.g. a global network of fuel cycle facilities) that a complete fuel cycle service can be provided. In addition, operations of the network's fuel cycle facilities would have to conform to international safety standards and be in a cost-competitive manner.

CONCLUSION

The global network of nuclear fuel cycle facilities proposed here is merely formed by a framework of contractual agreements among companies (and countries where companies are operating). The formation of the network is intended to provide full-scope fuel cycle services, which are economically competitive, meeting all applicable international safety standards, and complying with international safeguards and security requirement. Such a network does not need to be within a national boundary, and facilities in the network are not necessarily to be co-located in a nuclear fuel cycle center.

Such a network could relieve the "waste and non-proliferation" burden borne by current reactor operators/owners and certainly will be imposed on utilities/countries contemplating the development of new nuclear power programs. The formation of the network could reduce the incentive of countries acquiring their own fuel cycle capabilities (technologies and facilities), and consequently, may help reduce the overall safeguards and inspection costs. The network of fuel cycle facilities can also provide a "cradle-to-grave" fuel cycle service assuring a stable and reliable fresh fuel supply and managing the spent fuel and radioactive wastes in an environmentally sound manner.

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TECHNOLOGICAL OPPORTUNITIES TO INCREASE THE PROLIFERATION RESISTANCE OF GLOBAL CIVILIAN NUCLEAR POWER SYSTEMS

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Abstract

A special advisory Task Force to the US Department of Energy, known as the "TOPS" group, was requested to try to identify near and long-term technical opportunities to increase the proliferation resistance of global civilian nuclear power system and to recommend specific areas of research that should be pursued to further these goals. The Task Force has concluded that there are a number of promising areas of research and development that could be, and should be, pursued by the United States in collaboration with other countries that are likely to enhance the proliferation resistance of existing as well as potential advanced nuclear power systems. Three major subject areas have been recommended for ongoing US support:

- the development of improved methodologies to assess proliferation resistance of specific systems, including the evaluation of pathways other than the misuse of civilian nuclear power.
- the development and adaptation of technologies to strengthen application of institutional (or so-called, extrinsic) barriers against proliferation, (such as measures to help improve international safeguards).
- the development of new technologies to enhance the so-called "intrinsic" technical barriers of various systems against proliferation.

Several months ago, the Office of Nuclear Energy, Science and Technology, in the US Department of Energy, as well as DOE's Nuclear Energy Research Advisory Committee (NERAC) established a special Task Force. This group (which became known as the "TOPS" Task Force) was requested to identify both near and long-term technical opportunities to increase the proliferation resistance of global civilian nuclear power systems and to recommend specific areas of research that should be pursued to further these goals. The Task Force was also encouraged to recommend areas where international collaboration can be most productive. The membership of the group, which included sixteen people, was designed to represent a broad spectrum of viewpoints. The Chairman was John Taylor, of the Electric Power Research Institute in the US, and the author of this paper was a member of the group. Since this is an international subject, where foreign perspectives are important, representatives from foreign countries were invited to participate. In addition, the views of various research groups, industry, and technical organizations were solicited and integrated, as feasible, into the analysis. The Task Force operated on a consensus basis. While there were differences of view on the relative merits of different prospective technical courses of action, all Task Force members supported the basic recommendations in the group's report.

The Task Force concluded and so recommended to DOE that there were a number of promising areas of research and development (R&D) that could be, and should be, pursued by the United States in collaboration with other countries that are likely to enhance the proliferation resistance of existing as well as potential advanced nuclear power systems. It was recognized, however, that proliferation resistance was only one of the important components of improved nuclear power systems that are in need of further R&D. Other necessary steps are those that will advance the economy and safety of nuclear power systems as well as the ability of the nuclear industry to effectively manage nuclear wastes. It was the judgment of the Task Force that continued US participation in strengthening the global non-proliferation regime will depend, in part, on the preservation of US technological capabilities in the civil nuclear sector, including a strong US capability to carry out realistic and well-focused civil nuclear R&D. However, achieving and preserving this capability will require, both greatly increased government investment in forward-looking research and development, as well as the application of effective selectivity by the government and industry in deciding which of the several competing approaches to enhancing the nuclear option should receive priority.

In their deliberations, many members of the group felt that nuclear power has the potential to continue to make important contributions in meeting future global energy needs under conditions that are compatible with economic, nonproliferation, and environmental objectives, including efforts to abate air pollution and greenhouse gas emissions. It was observed that historically, the preferred approach for nations seeking nuclear weapons generally has been to establish a dedicated military program to produce the necessary nuclear material rather than by attempting to divert material from internationally safeguarded nuclear facilities. Nevertheless, it was clearly recognized that civilian nuclear activities can make direct or indirect contributions to the spread of nuclear weapons and some alleged civilian nuclear programs have provided a cover for military activities. Consequently, all Task Force members agreed that the continued exploration of new technical ways in which nuclear power systems can be made more resistant to proliferation should constitute an important ongoing feature in the improvement of the global non-proliferation regime. The Task force felt that the R&D to advance proliferation resistance should be designed to (a) assure that the utilization of civil nuclear power remains a comparatively unattractive route for those nations or groups interested in acquiring nuclear weapons, and (b) limit the degree to which the civilian nuclear energy system contributes to dedicated military programs.

To this end, the Task Force concluded that the United States (working with other countries) should support the exploration, and further development, of systems that:

- increase effectiveness of institutional or extrinsic measures (such as the IAEA safeguards system) that serve to underpin the entire international nonproliferation regime;
- make weapons-usable materials highly inaccessible (This should include an evaluation and as appropriate, the pursuit of advanced fuel cycle systems that avoid direct access to these materials);
- reduce the attractiveness of nuclear materials for potential weapons purposes;
- reduce the quantities of directly weapons-usable materials produced per unit of energy output; and
- can limit the spread of highly specialized knowledge and skills that can be directly used to design and fabricate nuclear weapons;

It was strongly recommended that the United States should evaluate, in close cooperation with others, a range of reactor and fuel cycle options that could potentially meet these objectives. However, these studies should not be conducted in isolation from other efforts to develop improvements and should be integrated with other efforts to assure that future nuclear power systems will be economical, safe, and environmentally friendly. It was noted that a number of comprehensive assessments had been performed in the past that addressed the non-proliferation characteristics of different nuclear power systems and it was urged that the US and other countries should draw from this extensive experience.

The Task Force noted that there are some decidedly different views within the international community as to how the nuclear fuel cycle and especially the "back end" can best be managed or whether this question simply should be deferred. Accordingly, the Task Force was of the view that different fuel cycles and reactor choices may continue to be followed by different nations. However, in all practicable cases, the group felt it would be desirable for the United States to be involved in cooperative R&D efforts with other nations and to have the technical ability to influence these programs so that they advance in ways that enhance proliferation resistance while also advancing economic and safety objectives. To this end, it was felt that a new US effort to pursue R&D at least initially at the conceptual level (and involving the conduct of analytical and experimental studies) that would evaluate and explore advances in proliferation resistance in different nuclear systems could strengthen the US ability to exert a constructive technical influence on future developments. More broadly, and for the longer term, for nuclear power to provide a significant fraction of the carbon-free energy the world is likely to need in the 21st Century, it was felt that the utilization of nuclear power would have to expand many fold. The realization of this goal, however, may be dependent, in part, on broad confidence in governments and publics that such an expansion will not significantly aggravate the proliferation problem. Thus, continued improvements in proliferation resistance, like continued improvements in nuclear safety, waste management, and economics could be important to the future growth of nuclear power.

Three major subject areas were recommended for ongoing support. Specifically, efforts should be made:

- to develop improved methodologies to assess proliferation resistance of specific systems, including the evaluation of pathways other than the misuse of civilian nuclear power.
- to develop/adapt technologies to strengthen application of institutional (or so-called, extrinsic) barriers against proliferation, such as measures to improve international safeguards materials protection, and control and accountancy.
- to develop new technologies to enhance the so-called "intrinsic" technical barriers of various systems against proliferation.

Since R&D will be critical in helping to make subsequent decisions on the appropriate paths to actually follow, it was felt the effective implementation of this proposed new R&D initiative will require a strategic planning approach that provides a basis for prioritization and subsequent selection of the desired longer-term R&D portfolio. At each significant step of R&D, the evaluation of the benefits/risks of new technical approaches and advanced systems has to take into account other significant objectives, including safety, environmental impact, economics, and waste management as well as proliferation resistance.

Framing and implementing the desired new R&D agenda also will require a systems perspective and an emphasis on comparative evaluation. The pursuit of most of the individual projects designed to improve barriers to proliferation should be carried out in the context of the overall development of the reactor or fuel cycle concept to which they are intended to apply and should address all the facilities of an integrated system so as to significantly reduce proliferation and national security concerns. Since several of the advanced concepts that one might choose from will take many years to commercialize, proliferation resistant improvements should be given significant attention in the early stages of development.

Establishing appropriate and realistic time frames for R&D is important. It was proposed that R&D programs should be established with three distinct time frames in terms of completion of the development and implementation of the technologies. The initiation of related R&D to be pursued in all three time frames would ideally start now, but selections will need to be made on the desired starting times based on the amount of available funding and following further screening, the priorities give to various programs. The time phases should include:

- Shorter-term projects likely to produce tangible results in about five year's time.
- Intermediate projects likely to produce tangible results up to about 15 years from now, and
- Longer-term projects. A commitment is critical to the longer-term exploration and, as appropriate and feasible, further development of advanced reactors and fuel cycles. However, nearer-term concrete needs should not be ignored in this process.

To provide tangible results that can affect proliferation resistance *in the first five years*, emphasis should be devoted to such areas as:

- Developing improved and standardized methodologies, including quantitative ones, for performing comparative assessments of the proliferation attributes and merits of different reactor and fuel cycle systems.
- Pursuing various nearer-term and concrete ways to strengthen the application of the extrinsic (or institutional) non-proliferation regime with emphasis on supporting international safeguards and national MPC&A programs; and
- Performing analytical studies and experiments designed to evaluate potential improvements in the intrinsic proliferation barriers for existing nuclear systems as well as potential advances in proliferation resistance in several advanced nuclear reactors and fuel cycle systems.

It was recognized that new technical efforts to strengthen international safeguards have to build on and be well coordinated with the national support programs for the IAEA safeguards systems that are already underway.

While the Task Force did not review in depth extensive ongoing safeguards R&D supported by the United States and other governments, a Special Working Group that included safeguards experts developed a list of potential areas where additional R&D in support of international safeguards and national MPC&A systems would be useful. These included ways to improve: (a) information technologies for safeguards; (b) safeguards system integration and studies (including integrating and balancing traditional and new safeguards measures); (c) material accounting and facility monitoring; (d) wide-area environmental monitoring; (e) material and item tagging; (f) safeguards cost-effectiveness; and (g) the integration of technological developments from a wide range of areas, including areas outside traditional nuclear science to advance safeguards.

Also, in the nearer-term, it will be important to pursue the evaluation of the adverse as well as positive implications that certain technological advances or deployments (such as those permitting production of weapons-usable material in smaller and more readily concealed facilities) might have for the global "extrinsic" non-proliferation regime.

In the initial five years, the initial emphasis in developing improvements in *intrinsic barriers* should be on examining ways to improve proliferation resistance in existing systems and assessing through analytic studies and experiments the potential inherent barriers that might be associated and pursued with the development of more advanced systems. For the first five years the primary focus would be on LWR "once through" systems e.g. achieving incrementally higher fuel burn-up.

In the intermediate period (of from about 6 to 15 years in the future), R&D themes should be explored or pursued that could lead to advances in the introduction of greater *intrinsic proliferation resistance in existing or future nuclear systems*.

Among the specific technical options for reactor and fuel cycle systems that have been proposed to improve proliferation resistance are:

- LWR fuel systems designed to produce smaller amounts of less attractive nuclear material in their spent fuel (such as higher burn-up, thorium-uranium [Th/U] fuels, and non-fertile fuels).
- LWR systems designed to allow recycle without separating weapons-usable material or providing facilities and processes that could not be readily modified for such separation (such as dry chemical reprocessing or recycle without reprocessing).
- High-temperature, gas-cooled systems designed so that the material in their spent fuel would be highly unattractive for weapons use.
- Liquid metal reactor and fuel cycle systems designed to avoid the production and separation of weapons-usable material, or the provision of facilities and processes that could be readily modified for such separation.
- Options for faster and more proliferation-resistant reductions in the world stockpiles of separated plutonium.
- Small modular reactor systems designed to offer a nuclear energy option with little potential for the host state to have access to weapons-usable materials and only very limited requirements and only very limited requirements for transfer of knowledge and technologies that could contribute to nuclear weapons programs.
- Transmutation technologies for spent fuel and nuclear wastes, which could reduce long-term safeguard requirements; and
- Dual-use advanced monitoring and analytical systems that can handle both safeguards needs and efficient plant operations, seeking improvements on systems already in place in countries like the United Kingdom and France.

It was recommended that the potential proliferation resistance of these various technological options should be evaluated and R&D should be pursued on those determined to be most promising and that would also meet other basic nuclear criteria (such as improved economics and enhanced safety) central to the DOE nuclear R&D program. The R&D on intrinsic barriers for particular systems that may be selected for support should be conducted from the outset as part of the overall development of such systems.

To provide tangible *results that can improve proliferation resistance over the longer term (16 years out*), it was recommended that projects should focus on the further evaluation, and, as appropriate, more active development, possibly through pilot plant or demonstration projects, of selected advanced systems and concepts. These efforts should consider and assess advanced light water reactors, liquid metal reactors, liquid-fuel reactors, and gas-cooled reactors. Various size reactor concepts should be investigated that do not require refueling for

10 to 15 years, with a realistic emphasis upon reducing dependence on high quality human support. Advanced closed fuel cycle options also should be investigated when they offer potential opportunities for improving proliferation resistance and international security. This should include the examination of systems that would avoid the presence of separated plutonium and HEU and of facilities and processes that could readily be adapted to produce such materials.

In conclusion, the Task Force suggested that DOE should budget an additional $25 million for these purposes starting, preferably in the US Fiscal Year 2002, increasing this level in subsequent years, if promising new opportunities are identified.

It remains to be seen what impact these recommendations will have on DOE and also the Congress. The DOE staff reacted very positively to these recommendations, but was unable to accommodate them in the proposed DOE budget for FY 2002, due to heavy pressures from The White House Office of Management and Budget to curtail increases to help accommodate the President's proposed tax cut. However, there are some strong personalities in the Congress, who wish to see more funding given to nuclear R&D and it is still conceivable, in my personal view, that some of these proposals from the TOPS group may well be accommodated in some manner.

NON-PROLIFERATION ASPECTS OF THE PBMR FUEL CYCLE

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Abstract

The paper presents a brief technical description of the Pebble Bed Modular Reactor (PBMR) design. This technical description is presented in sufficient detail in order to serve as a basis for understanding the design of the fuel handling system and the reactor core. The fuel fabrication process is also briefly described.

In the section describing the core design and associated reactor physics, the lack of ability of the reactor to produce nuclear material for use in nuclear explosives is presented. It is shown that, with reference to plutonium production, the mixture of isotopes produced during normal commercial operation is unsuitable for use in nuclear explosives.

The approaches, anticipated to be proposed by the International Atomic Energy Agency (IAEA) for the implementation of nuclear material safeguards in the fuel fabrication plant and reactor are briefly discussed.

The present status regarding high-temperature graphite moderated reactor reprocessing technology is briefly summarized.

Finally, a summary is presented of the non-proliferation attributes of the PBMR fuel cycle.

1. INTRODUCTION

The PBMR Module power conversion is based on a closed cycle circuit, utilizing a single loop direct gas cycle system that utilizes a helium cooled and graphite-moderated nuclear core assembly as a heat source. The coolant gas transfers heat from the core directly to the power conversion system consisting of gas turbo-machinery, a generator, gas coolers and heat exchangers. The reactor has a thermal power of 268 MW with an electrical output of 110 MW.

2. TECHNICAL DESCRIPTION OF THE REACTOR

The Reactor Pressure Vessel is approximately 6.2 m in diameter, approximately 20.5 m high, and manufactured from reactor grade forged steel with a wall thickness varying between 120 mm and 220 mm. It has an internal steel core barrel with an internal diameter of 5.8 m, and a wall thickness of 50 mm. This internal core barrel in turn supports the graphite reflector and carbon thermal shield. The combined radial thickness of the graphite and carbon is 1 m. The graphite reflector has 35 vertical borings to house the reactivity control units. The volume inside the graphite reflector has a diameter of 3.5 m and an effective height of 8.5 m. This volume is filled with the core, consisting of fuel in the form of 60 mm diameter fuel spheres, and a central reflector column of graphite spheres with the same diameter as the fuel spheres.

When fully loaded, the core would contain 310 000 fuel spheres and 110 000 graphite spheres. Helium coolant enters the reactor vessel at a temperature of approximately 500 ºC and a pressure of 70 bar. The helium flows down through the core, picking up the heat generated by nuclear fission, and exits the bottom of the reactor vessel at a temperature of approximately 900 ºC.

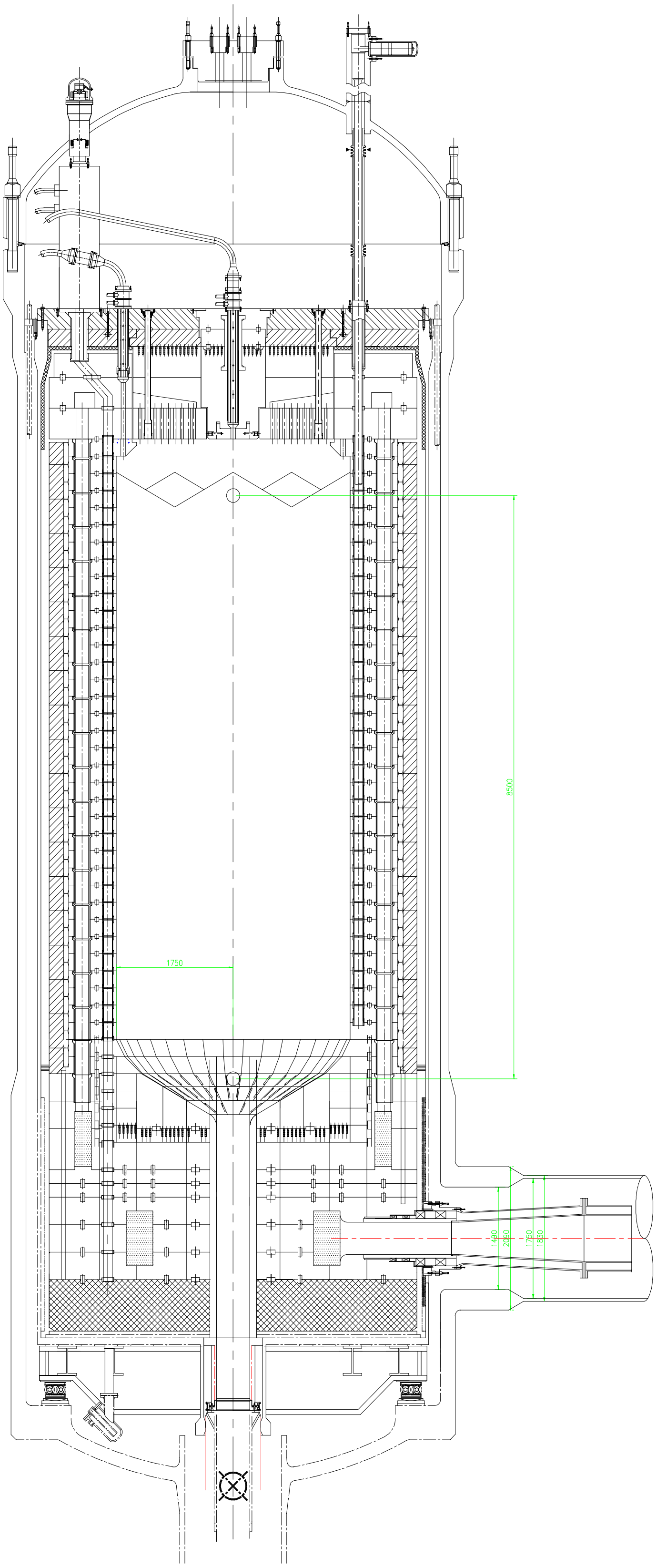

FIG 1: PBMR Core Structures layout within the Reactor Pressure Vessel

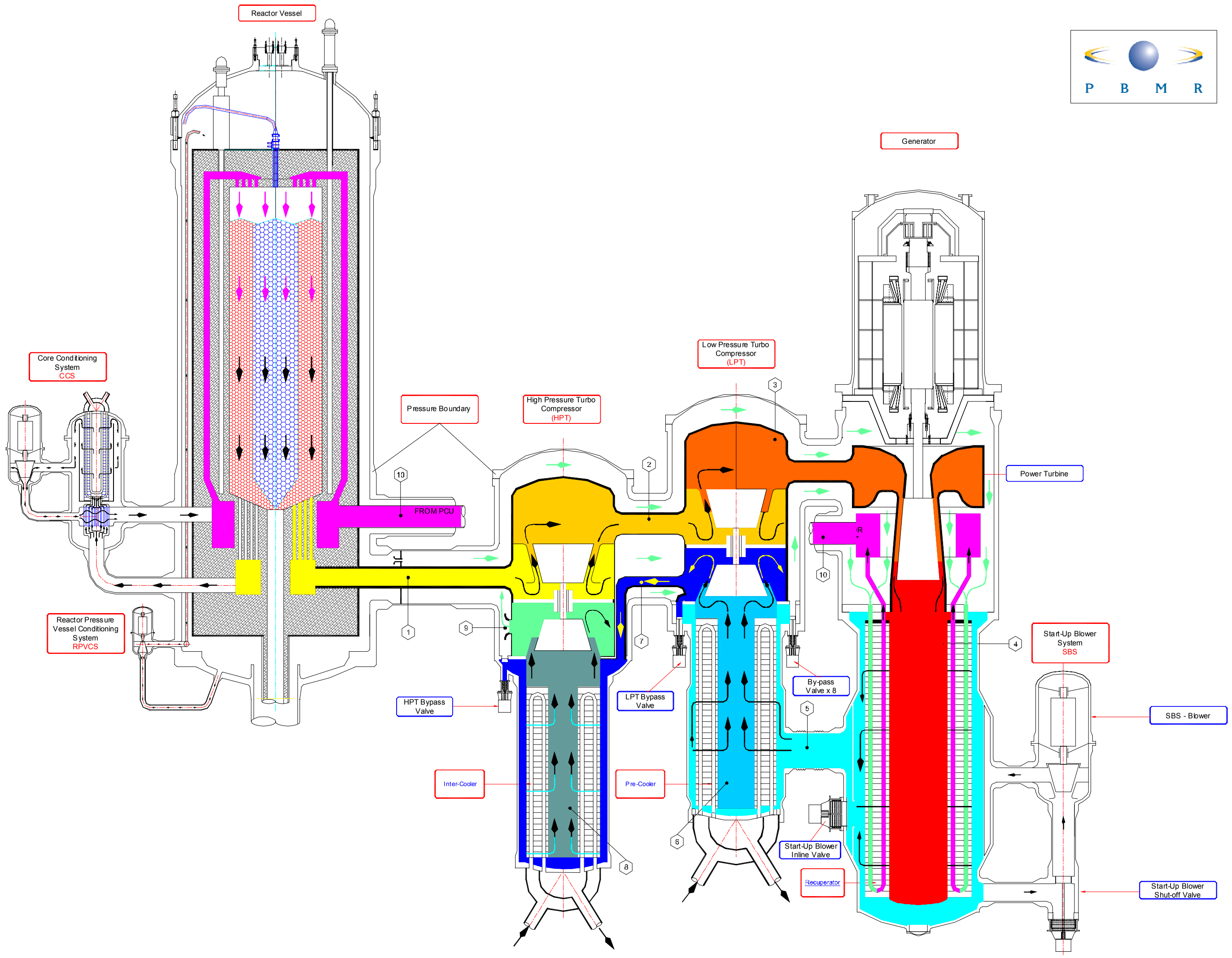

FIG 2: PBMR Main Power System (semi-chematic)

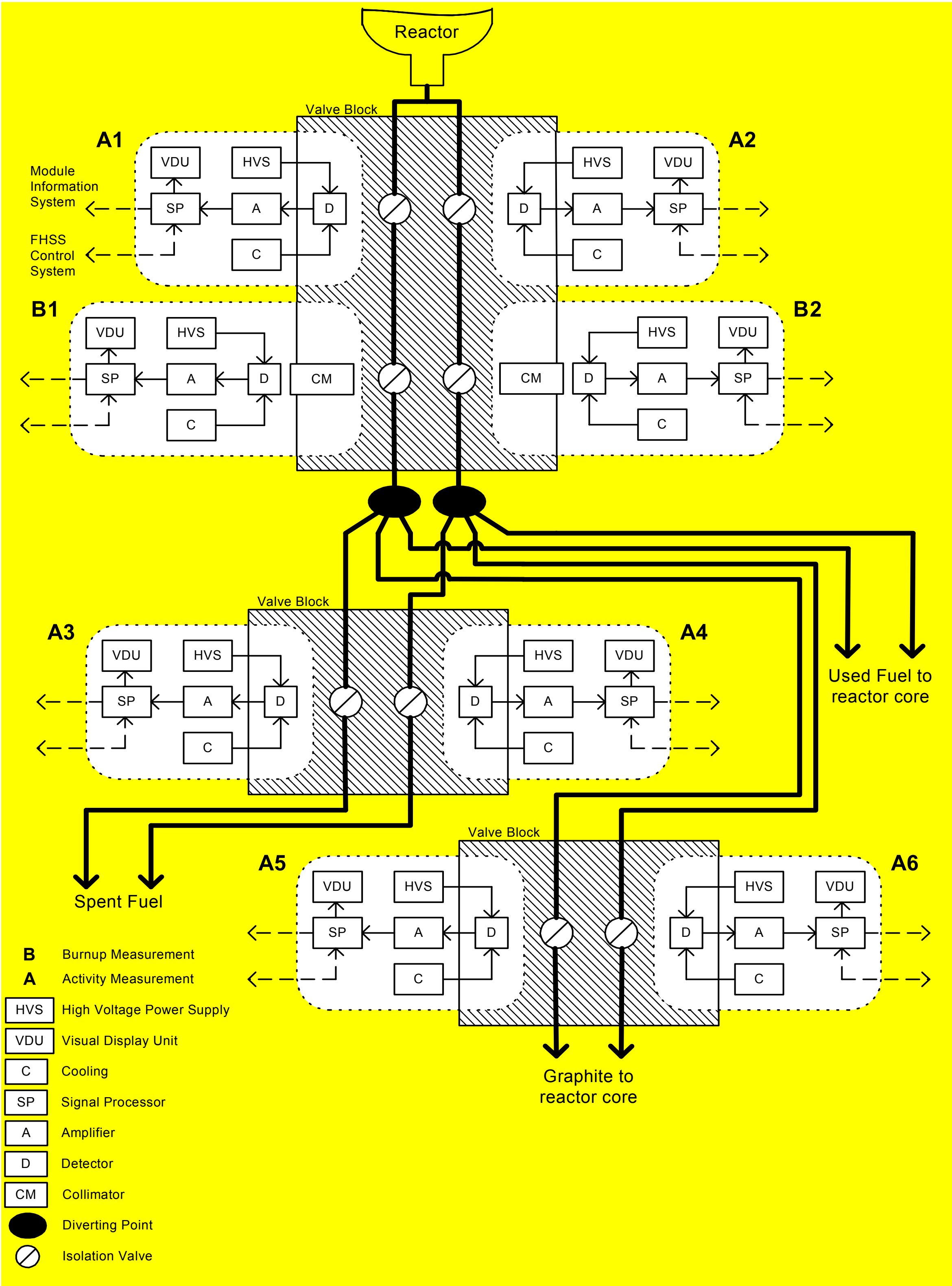

FIG 3: Simplified Fuel and Graphite Sphere Flow Schematic incorporating the associated Control and Instrumentation

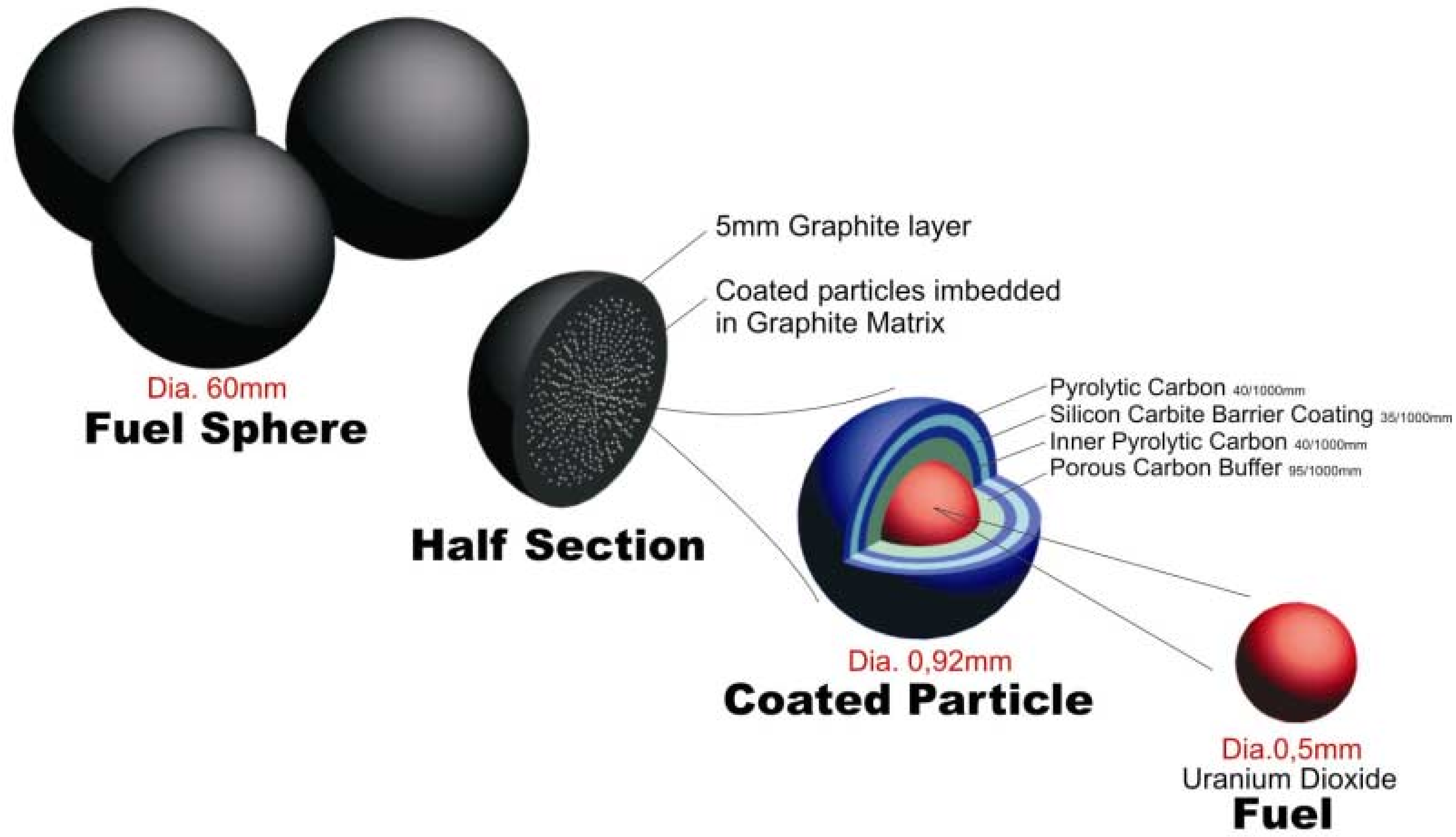

FIG 4. PBMR Fuel element design

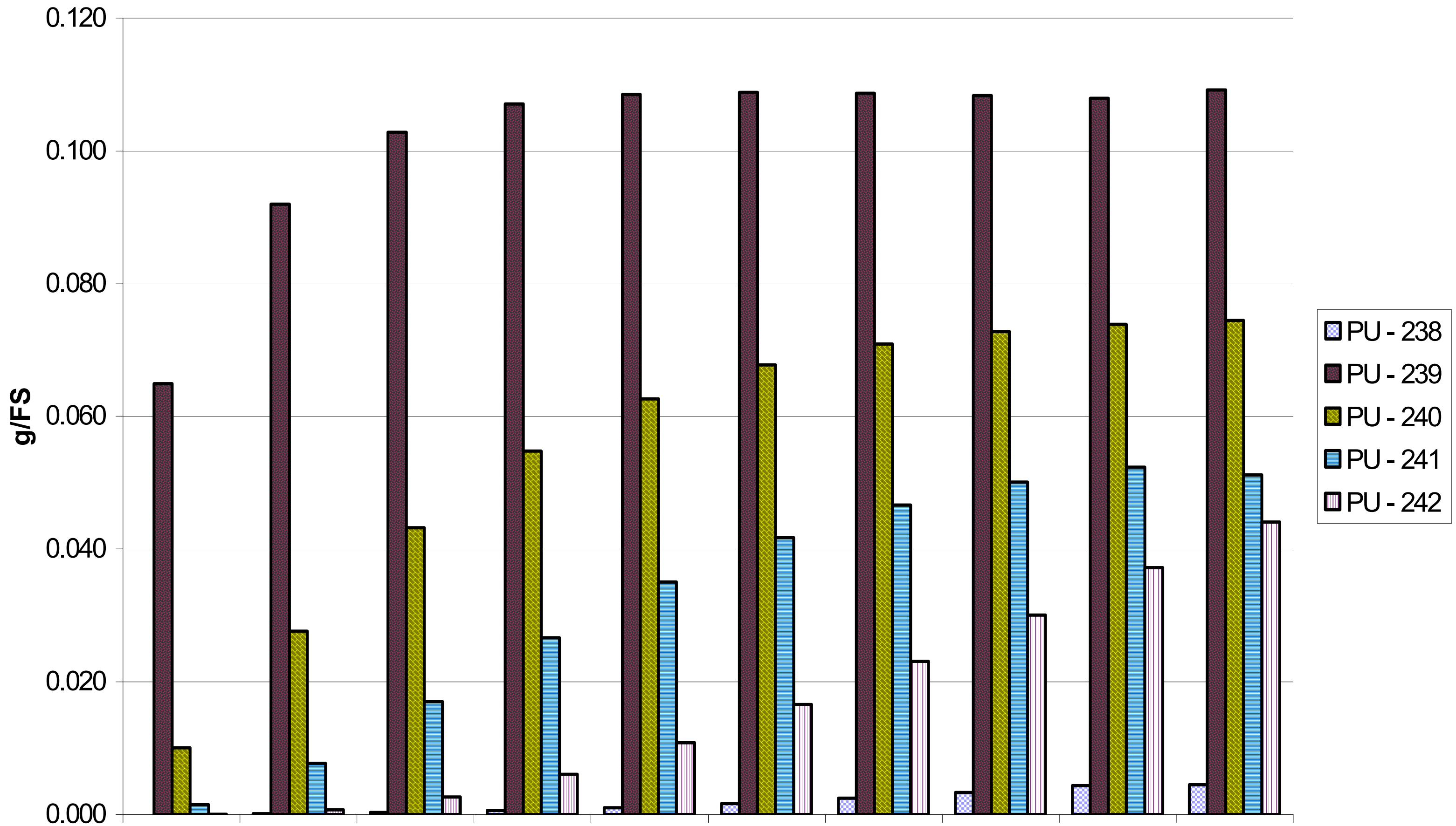

FIG 5. PBMR Pu Inventory versus the number of passes

The hot gas exiting from the core then enters the first of the three gas turbines in series. The first two turbines drive compressors, and the third turbine drives the electrical generator. The coolant leaves the last turbine at a temperature of approximately 530 ºC and at a pressure of 26 bar, after which it is cooled and recompressed to 70 bar, reheated to 500 ºC in a recuperator, and returned to the reactor vessel to re-enter the cycle described above. The reactor vessel is shown in Fig 1. Fig 2 presents the Main Power System and helium flow paths in a semi-schematic form.

An on-line fuel and graphite sphere loading and unloading scheme is used. The loading and unloading is done with a support system that has the following functions:

- Loading of the core cavity with graphite spheres.
- Loading of new fuel spheres into the core.
- Removing erroneously discharged fuel spheres from the graphite sphere system.
- Preventing erroneously discharged graphite spheres from initiating the loading of new fuel spheres, via radiation sensors fitted to the delivery line to the spent fuel storage tanks.
- Removing fuel and graphite spheres from the discharge tube.
- Separating damaged spheres.
- Recirculating graphite spheres.
- Recirculating partially used fuel spheres through the core.
- Measuring burn-up of partially used fuel spheres, and discharging spent fuel spheres into the spent fuel storage system.
- Replacing graphite spheres from the graphite storage buffer tank due to broken graphite spheres.
- Defuelling and refuelling of the core, by transfer of the core inventory from the reactor into separate graphite and fuel storage tanks located in an area adjacent to the reactor, during maintenance intervention requiring the venting of the main power system to atmosphere.
- Reloading the core from these tanks during refuelling of the core.

Fig 3 presents a simplified block diagram of the fuel and graphite sphere flow schematic and associated Control and Instrumentation required for the routing of the fuel or graphite spheres to their appropriate destination during normal operation.

3. THE FUEL ELEMENT

3.1. The Fuel Element Production Process

3.1.1 Production of kernels

Starting from U_3O_8-powder, this powder is dissolved in nitric acid forming uranyl nitrate. After organic agents have been added, and the solution neutralized with ammonia, the solution flows through oscillating nozzles, thus being transformed into droplets.

Falling through a gaseous ammonia atmosphere, the droplets will synthesize a tenacious surface before dipping into an aqueous ammonia solution. In the solution, these small droplets gel by forming ammonium uranate. Subsequently, the droplets are aged to improve the internal structure, and then washed to remove the ammonium nitrate and organic additives. This is followed by drying, calcining, reducing to UO_2 with hydrogen, and finally, sintering to produce kernels of 99% theoretical density.

3.2. Production of Coating

Coating is done by passing a stream of carrier gas (argon and or hydrogen) upward through a batch of kernels, and in so doing, creating a fluidized bed. The bed is kept at a controlled high temperature in a gas-tight furnace. Different organic gases are added to the fluidizing gas in subsequent steps. The decomposition of these organic gaseous compounds forms the desired coating layers. For the buffer and the pyrolytic carbon (PyC) layers, the agents are hydrocarbons, and for the silicon carbide, the agent is methyl trichlorosilane.
The different layers are deposited consecutively without interruption of the process, or discharging until all the layers have been deposited.

3.2.1 Fuel element fabrication

The raw materials used in the fabrication of the fuel element are graphite powder and phenolic resin, processed to give a homogeneous mix of resinated graphite powder. The graphite is a mixture of natural and artificial electro-graphite powder. The mixture of natural and artificial electro-graphite powder is chosen to obtain relatively isotropic properties of the fuel sphere before and after irradiation.

The coated particles are then filled into a portion of the resinated graphite powder, thoroughly mixed, and pre-moulded into the fuelled part of the sphere that is approximately 50 mm in diameter. Thereafter it is pressed into a mould made of silicone rubber, which is partially filled with resinated graphite and cold isostatically pressed.

The final steps in the fabrication process are lathing to the desired diameter, followed by a heat treatment step for coking of the resin and removal of impurities.

3.3. The Fuel Element

Fig 4 shows the design of the fuel element used for the PBMR.

4. DESCRIPTION OF CORE DESIGN

The equilibrium core of the PBMR is filled with approximately 333 000 fuel elements and 110 000 graphite spheres in the central reflector zone. Each fuel element contains 9 g of uranium having an enrichment of approximately 8.1%.

The design caters for a target burn-up of approximately 80 000 MWd/t U. When this burn-up is reached, the total fissile content is approximately 2.0% by weight, 1.3% of which will be U-235.
The fuel is circulated on average 10 times through the core, resulting in a variation in average burn-up from top to bottom of the core of only 8 000 MWd/t U.

The core has 18 control rods. Nine of the control rods are used for reactivity control during load changes, and the other nine are used to shut down the reactor. The reactor design caters for a load following capability within the range 100–40–100%. The nine control rods will be inserted to a depth during equilibrium operation to provide reactivity compensation for xenon poisoning effects during load following operation.

In addition to the control rods, an additional shutdown system is used to shut the reactor down to cold conditions. This system utilizes small absorber spheres, which are dropped under gravity into 17 borings in the reflector. Removal of these spheres is by means of a helium gas conveyance system.

A summary of the characteristic core data is provided in Table 1.

TABLE I: CHARACTERISTIC DATA FOR PBMR CORE OPERATING AT 268 MW THERMAL

Parameter	**Units**	**Value**
General data		
- average enrichment in the core	%	4.55
- average fuel residence time	Days	874
- average burn-up	MWd/T	80 000
- conversion ratio*		0.452
- power peaking maximum/average		5.66
- maximum power per fuel sphere	KW	
In fuel zone		1.67
In mixing zone		2.34
Average thermal neutron flux	$n/cm^2.s$	1.03×10^{14}
Average total flux	$n/cm^2.s$	1.82×10^{14}
Reactivity values		
- shutdown rods	$\Delta k_{eff}/ k_{eff}$	0.043
- control rods	$\Delta k_{eff}/ k_{eff}$	0.034
- small absorber spheres	$\Delta k_{eff}/ k_{eff}$	0.054

$$\text{* Conversion ratio} = \frac{\text{Production rate of fissile nuclides}}{\text{Rate of loss of fissile nuclides}}$$

The average core plutonium inventory as a function of the number of fuel element passes through the core is shown in Fig. 5.

5. NUCLEAR MATERIAL SAFEGUARDS APPROACHES

5.1. Safeguards at the PBMR Fuel Fabrication plant

The application of nuclear material safeguards at the fuel fabrication plant will utilize the measures currently employed by the IAEA at similar bulk nuclear material handling facilities. Briefly stated, these measures are:

- weighing and non-destructive assaying (NDA) of feed and intermediate product materials;
- weighing and sampling of feed and intermediate product materials for the purpose of performing destructive analysis (DA) for element and isotope specific determinations; and
- weighing and NDA of finished fuel spheres.

IAEA seals will be applied on containers of fuel spheres before shipment to the facility.

5.2. Safeguards at the PBMR

5.2.1 Fresh fuel storage

The fresh fuel will be stored in a fresh fuel storage room. The fuel will be stored in 70 containers, each containing approximately 1 000 spheres. This will represent an equivalent of six months' fresh fuel supply.

5.2.2 Spent fuel

The fuel will be circulated through the core, and by performing a burn-up measurement after discharge from the core, usable fuel will be recirculated back into the core. After the fuel has reached an average predetermined burn-up, which in the present design will occur after approximately 10 passes through the core, it will be treated as spent fuel, and will then be routed to the spent fuel storage tanks. As shown in Fig. 5, it is expected that a spent fuel sphere will then contain an amount of 0.3 g of total plutonium isotopes.

5.2.3 Spent fuel storage

The spent fuel will be stored in spent fuel tanks. The storage capacity is such, that at the end of facility life, a total equivalent of 10 core loads of spent fuel will be stored in this facility. It is envisaged that during the operational lifetime of the reactor, i.e. 40 calendar years, no spent fuel will be shipped out of the facility, and furthermore, that it can be stored in the spent fuel storage after final shutdown, for a further period of 40 calendar years.

5.2.4 The reactor core

As described earlier, the reactor core will contain a total of 330 000 fuel spheres and 110 000 graphite spheres with fuel loaded and unloaded with the reactor at power.

5.2.5 Nuclear material inventories expressed in Significant Quantities

The term Significant Quantity (SQ) is defined [1] for the purpose of this document as follows:

- low enriched material (U-235 content) 75 kg
- plutonium (all isotopes) 8 kg

Using this definition, the following inventories expressed in Significant Quantities are expected to be present following 40 years of operation:

Fresh Fuel Storage	0	SQ of low enriched uranium
Reactor Core	2	SQ of low enriched uranium
	8	SQ of plutonium
Spent Fuel Storage (end of reactor life)	5	SQ of low enriched uranium
	124	SQ of plutonium

It should, however, be noted that the spent fuel inventory will build up to the end-of-life value over a period of 40 years, with the average inventory of three SQ of total plutonium isotopes being added each year. The fresh fuel storage will contain a nominal quantity of 0.68 SQ of low enriched uranium during the operational lifetime of the reactor.

5.3. Anticipated nuclear material verification activities

Fresh fuel verification will be done using conventional non-destructive assaying. However, once the fuel is loaded into the core and fission products are produced, direct access to the fuel becomes impossible. It is anticipated that the core will possibly be designated by the Deputy Director General, Department of Safeguards (DDG-SG) of the IAEA, as Difficult-to-Access, in which case the access routes or openings to or from the core will have to be sealed, and surveillance cameras will be used to provide a dual function.

To some extent, the same is true for the spent fuel storage. However, this system could be engineered to provide an IAEA authenticated fuel counting system for the spent fuel discharged into the spent fuel tanks. In addition, a facility could be established to detect the presence of plutonium in a randomly selected tank by means of gamma spectroscopy. The detector of such a system could be inserted into a special access tube. One tube would be provided for each tank, and when not in use, these tubes could be sealed with an IAEA sealing system.

6. STATUS OF HTR FUEL REPROCESSING TECHNOLOGY

Methods for reprocessing High Temperature Reactor (HTR) spent fuel elements have been developed in the past. These methods incorporated combinations of thermal, chemical and mechanical processes. However, none of these processes has ever reached a large-scale (commercial) status, both for political and economical reasons. Since reprocessing is technically feasible, it could become the method of choice in the event of limited supply of uranium and/or thorium.

7. NON-PROLIFERATION ATTRIBUTES OF THE PBMR FUEL CYCLE

- Low enriched fuel is used.
- A closed system for fuelling and defuelling with on-line tracking of fuel or graphite sphere location. This reduces the possibility of clandestine introduction of target material, or the protracted diversion of core nuclear material.
- Only sufficient excess reactivity is allowed to cater for temperature effects and to provide for equilibrium and transient fission product poisoning. This also reduces the possibility of clandestine introduction of target material.

- The PBMR is designed to store all the spent fuel generated during the operational lifetime of the reactor in the facility.
- The reprocessing technology for HTR fuel has never reached a mature status.
- Should reprocessing become a viable option, the high burn-up achieved by the fuel, produces a mixture of plutonium isotopes that does not favour the production of a reliable nuclear explosive device [2]. In addition, the heat generated by these isotopes will cause rapid degradation of the high explosive components of such a device.

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[1] IAEA Safeguards Glossary, IAEA/SG/INF/1 Rev. 1, 1987.

[2] J. CARSON MARK, Explosive Properties of Reactor-Grade Plutonium Science and Global Security, 1993, Volume 4, pp 111-128.

BUILDING PROLIFERATION-RESISTANCE INTO THE NUCLEAR FUEL CYCLE

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Abstract

The nuclear non-proliferation regime rests on several elements that complement and reinforce each other. The political commitment of States against possession of nuclear weapons is reinforced by institutional measures, the most important being IAEA safeguards. The institutional barriers can be effectively reinforced by technological barriers against proliferation. At the very least, technological barriers could make breakout from the non-proliferation regime more difficult and time-consuming, thus providing enhanced deterrence and better opportunity for the international community to intervene. Under an integrated safeguards regime – a regime that optimally combines safeguards measures from comprehensive safeguards agreements of INFCIRC/153-type and the Additional Protocol (INFCIRC/540) – it would be possible to give greater weight to the technological barriers to proliferation. Under such a system fuel cycles that are inherently proliferation resistant could be expected to produce significant benefits in terms of reduced safeguards effort.

1. INTRODUCTION

Within the broader energy market there is increasing recognition that small to medium-sized power units, placed close to regional demand centres, are a useful supplement to large centralised power generation units. The higher "per energy unit" cost of energy of the smaller units can be offset by lower transmission and attendant transmission infrastructure costs. In the recent past these smaller distributed generating units have principally been fuelled by natural gas, but as the price of natural gas is rising rapidly, the true cost of such generation is being realised. This is leading to reconsideration of smaller, modular nuclear generation modalities with the potential for lower overall costs. The use of smaller units, distributed among regional demand centres can be expected to result in structurally robust energy markets which are not prone to the supply shortfalls that occur during the outages of large centralised generating units.

Small and medium-sized nuclear reactors can be used to complement large nuclear power units by supplying electricity, heat and desalinated water to remote areas. In the wrong hands, however, these reactors might become a means towards proliferation of nuclear weapons. The nuclear non-proliferation regime allows States to have confidence that their trading partners, neighbours and other fellow Treaty signatories are complying with their non-proliferation commitments. It helps to ensure that material within civil nuclear cycles is used for exclusively peaceful purposes.

The nuclear non-proliferation regime rests on several elements that complement and reinforce each other. The political commitment of States against possession of nuclear weapons is reinforced by institutional measures, the most important being IAEA safeguards, which provide a high level of assurance of compliance with obligations through international verification.

It has been argued by the authors [1-2] (and many others) that the political commitments and institutional barriers against proliferation, such as treaty regimes and associated verification arrangements, can be effectively reinforced by technological barriers. At the very least, those barriers could make breakout from the non-proliferation regime more difficult and time-consuming, thus providing enhanced deterrence against diversion and better opportunity for the international community to intervene should a State be found to be in breach of its commitments.

With the introduction the Model Additional Protocol (INFCIRC/540) and the move towards an integrated safeguards system, technological barriers to proliferation can be given additional weight in establishing a system of safeguards to be applied to a State. For the State this could have the benefit of lowering the overall intrusiveness of the international safeguards inspection regime while still allowing the State to demonstrate its compliance with its international commitments. For the IAEA it could have the benefit of slowing the growth in inspection effort and associated costs, allowing effort to be concentrated in areas of the fuel cycle of greatest proliferation concern. Reducing the costs of safeguards has benefits for all Member States of the IAEA as it lowers the contributions that currently support the safeguards effort.

Starting with a discussion of the strategic value nuclear material and reactor-associated fissile material acquisition paths, we discuss three basic approaches to enhance proliferation resistance of small and medium-sized reactors, namely: (1) reduction of strategic value of materials involved in nuclear power generation; (2) incorporating reactor design features preventing diversion of material; and (3) facilitating safeguards implementation.

The views contained in this paper are the views of the authors and not necessarily the views of the Australian Government.

2. THE STRATEGIC VALUE OF NUCLEAR MATERIAL

The strategic value of any particular form of nuclear material is determined by the degree of difficulty that would be experienced in converting the material into a weapons-useable form. Materials that are used or stored in a form suitable for weapons have the highest strategic value.

2.1. Weapons-Useable Material

The manufacture of nuclear weapons requires either:

- pure uranium metal at very high enrichment levels (though the HEU category starts at 20% U-235, *weapons-grade* uranium comprises 93% or more U-235), produced in enrichment plants designed and operated for this purpose; or
- pure plutonium metal preferably with a very high proportion of Pu-239 (*weapons-grade* plutonium comprises less than 7% Pu-240), produced in reactors designed and operated to produce low burn-up plutonium, and separated from spent fuel or irradiation targets in reprocessing plants or plutonium extraction plants.

These weapons-useable materials are very different to those normally produced in civil programs:

- low enriched uranium (LEU) typically used in light water reactors (LWRs) is in the range of 3-5% U-235. The utilisation of LEU as a source material for weapons would require chemical, enrichment and metallurgical processes, increasing the time frame for the production of weapons-useable material significantly compared to the use of HEU as the source material;

- reactor-grade plutonium (RG-Pu) from the operation of LWRs is of around 25% Pu-240 or higher. Any attempt to utilise RG-Pu for weapons would encounter substantial technological challenges compared to the use of weapons-grade plutonium.

2.2. Material Features Affecting Its Strategic Value

The *isotopic composition* of the material intended for the use in weapons could be an efficient barrier to proliferation as it directly relates to the relative difficulty of: manufacturing a nuclear weapon with material of a specific isotopic composition; or altering its isotopic composition to obtain weapons-useable material. In other words, materials with a higher isotopic proliferation barrier would require more advanced (and thus hopefully less available) weapon designs and technology for their processing into weapons-useable form.

Attributes that are important for determining the effectiveness of the isotopic proliferation barrier and which need to be taken into account when designing and manufacturing a nuclear device include:

- the critical mass of material (an attribute directly associated with its isotopic composition);
- the spontaneous neutron generation rate that might complicate design, and affect a weapon's yield and reliability;
- the heat and radiation outputs of the material.

The *chemical form* of material can also serve as a proliferation barrier. This relates to the relative effort required to: refine materials into the appropriate form; or chemically process fissile material to separate it from accompanying diluents, contaminants or any other admixtures that might be incorporated to frustrate chemical separation; in order to obtain materials of sufficient purity for weapons applications.

The chemical barrier effectiveness of some of the more common materials involved in the nuclear fuel cycle can be roughly classified in the following order (from simplest to most difficult): pure metals, conventional compounds (e.g. oxides, nitrides), mixed compounds (e.g. fresh MOX fuel), spent fuel, non-conventional compounds (e.g. carbides and silicides), and vitrified wastes (borosilicate glasses and titanium oxide forms).

3. FISSILE MATERIAL ACQUISITION PATHS ASSOCIATED WITH REACTORS

There are a variety of paths available for States that might wish to acquire fissile material in violation of their international commitments. One of the most important reasons for the existence of the international safeguards regime is to have the capability to detect such violations and to deter them by placing an element of risk that the acquisition would be detected in a timely fashion. In order for there to be an appreciable risk of detection, the IAEA has to consider each plausible acquisition path and introduce measures to deal with that path in an appropriate way.

If the Agency devotes a great deal of resources to addressing one particular material acquisition path at a facility but ignores others, then the overall result will be less than satisfactory. The Agency must perform a thorough "diversion path analysis" and tailor the implementation of its safeguards efforts to address the real risks of diversion.

3.1. Diversion of Unirradiated Direct-Use Material

There are many nuclear facilities in the world that have material that – for safeguards purposes at least – is considered to be in a form directly useable by would-be proliferators. Such material is generally referred to as *Unirradiated Direct-Use Material* (UDU). This description is applied to high enriched uranium (HEU – containing 20% or more U-235), uranium-233 and plutonium (of almost any isotopic composition) regardless of their chemical form.

Such material can be found as fresh fuel at Materials Testing Reactors (MTRs), Research Reactors (RRs), Critical Assemblies (CAs) and any facility which is using HEU fuel, Mixed Oxide (MOX) fuel or any other plutonium or U-233 fuel. UDU is the most sensitive and closely controlled material in the international safeguards system.

There are many possible ways for a State to divert UDU material – the most obvious (and the most difficult to counter) is described as a *"crash through" approach*. Under this scenario a proliferator would simply take the material from its safeguarded storage area as soon as the IAEA inspector had finished performing one inspection. The intention would be to have processed the material into a form suitable for use in a weapon before the next inspection falls due. At this point the proliferator could declare itself to be in possession of a nuclear weapon (or weapons) and the whole world would know that it was in breach of its safeguards obligations.

There are also certain *less dramatic scenarios* for the acquisition of UDU for a State with facilities containing material of that type. For example the operator could replace one or more items either with inactive dummies or with dummies which in some way mimic the material taken (such as borrowing equivalent material from another facility within the State). The aim would be to take the risk that the statistical sampling plan applied to the population of fresh fuel assemblies by the IAEA would fail to note the substitution. An alternative is to take small amounts of material from many items. The expectation would be that the small loss from many items would be within the statistically accuracy limits of the measurement system used by the IAEA during the inspection and consequently the overall diversion would be undetected.

Other acquisition paths for UDU include *the undeclared import of the material or manufacturing the material from undeclared source material using indigenous enrichment technology*. Under the classical safeguards system, formal consideration was only given to the paths that involved acquisition from declared sources – with the advent of the Additional Protocol, measures are increasingly in place to deal with acquisitions from any source – not just declared sources.

The acquisition of fissile material from fresh fuel is a relatively straightforward exercise and it is its very simplicity that makes it so difficult to prevent. If a facility has a sufficient quantity of UDU material the IAEA will generally conduct inspections on a monthly or biweekly basis. If facility conditions make it practical, a large part of the inventory will be covered by containment or surveillance measures and the remaining inventory will be subject to frequent re-measurement. The aim is to provide a heightened level of deterrence by ensuring that any diversion would be detected in a short enough interval that even a "crash through" scenario is unlikely to be successful before it is detected.

3.2. Diversion of Irradiated Direct-Use Material

Material that has been irradiated in a reactor normally has a high output of heat and radiation and requires heavy shielding and special tools to be handled or processed. Because of these special factors it is acknowledged that *acquiring material suitable for weapons from Irradiated Direct-Use Material (IDU) is much more complicated than a similar acquisition from UDU.*

To acquire fissile material from the declared irradiated fuel from a reactor, a proliferator would need to take either an adequate number of complete spent fuel assemblies or a very large number of irradiated fuel pins from a large number of assemblies. This material would need to be transported away from the reactor in heavily shielded casks in order to deal with both the heat and radiation generated by the assemblies or pins. The reprocessing of the spent fuel or irradiated pins has to take place behind massive shielding and all of the necessary equipment must be operated remotely.

A "crash through" scenario for IDU material involves diverting the material immediately after an IAEA inspection, but unlike the case for UDU, the material must be reprocessed before it can be used for weapons. Reprocessing appreciable quantities of spent nuclear fuel and producing UDU from IDU is not something that can be accomplished very quickly. UDU can theoretically be processed into weapons components in a matter of days, while, even under the best of circumstances it would take some months to process IDU to produce UDU.

There are many possible diversion scenarios for spent fuel, but as all of these scenarios require the special handling equipment and extensive shielding that were mentioned earlier, there are relatively simple measures that can address a whole range of diversion scenarios.

Smaller reactor facilities generally have smaller fuel assemblies with lower fuel loadings per assembly – however, in general these factors do not greatly simplify the tasks that must be undertaken by a would be proliferator. Spent fuel from small power reactors, MTRs and the great majority of RRs is intensely hot and radioactive and requires comparable levels of shielding to large power reactor fuel in order to be handled safely.

In general, acquisition of IDU from small power reactors is much more complicated than an equivalent diversion from an MTR or RR. MTRs and RRs generally have means to introduce items into neutron beam lines or other irradiation stations. As these items also require the heavy shielding that is required to transport spent fuel they would provide a regular cover for potential diversion activities.

The IAEA considers all of the plausible "acquisition paths" or "diversion scenarios" in establishing a safeguards approach for a facility. The degree of difficulty inherent in the acquisition path is assessed, as well as the time required for successful completion. Where engineering controls have been established that limit the possibility for the successful completion of a particular diversion scenario it is possible to take account of this in establishing the safeguards approach (these engineering limitations will be discussed later in this paper). The frequency and intensity of inspection effort is set to ensure that every reasonably achievable acquisition path is covered by appropriate safeguards measures.

Most commonly, this involves inspections at regular intervals with either some form of verification activity or with the review of some form of containment and surveillance measures to ensure that continuity of knowledge on the spent fuel items has been maintained.

At power reactors in countries subject to the new Integrated Safeguards regime, current plans are to remove surveillance measures from the spent fuel pond area and rely on annual reverification of the spent fuel as the major safeguards measure. This practical step is being taken in countries in which the IAEA has been able to derive credible assurance as to the absence of undeclared facilities and activities. The fissile material in spent fuel is accessible only after reprocessing and the assurance that there is no undeclared reprocessing capability within a State makes unnecessary the current quarterly inspections for spent fuel.

3.3. Undeclared Irradiation

IDU material can also be produced at a range of nuclear facilities by irradiating fertile material in the neutron flux of the core. Plutonium can be bred from natural or depleted uranium and uranium-233 can be bred from thorium. The degree to which this is a realistic acquisition path depends heavily on the power output of the reactor and on the configuration of the reactor core. In the case of MTRs and RRs it has been calculated that in order to produce 8kg of plutonium or uranium-233 within a twelve month period a reactor with a thermal power rating of at least 25 MW would be required [3]. A similar minimum power level would apply to small power reactors. For any power reactor with a thermal power output greater than 25 MW (which is effectively all power reactors), some consideration must be given to addressing the possibility of unreported fissile material production.

Unreported fissile material production is a difficult acquisition path to cover for MTRs and RRs (most especially those with thermal power outputs in excess of 25 MW). The purpose of such reactors is generally to gain access to the neutron flux on a regular basis – such activities are entirely legitimate but they would also provide the perfect cover for covert acquisition of IDU.

In general, small power reactors present fewer possible acquisition paths for the undeclared production of fissile material than MTRs and RRs. As the principal purpose of a power reactor is to produce power (or in special cases, heat and/or desalinated water) rather than neutron beams there are, in general, greater complications involved in using such a reactor for unreported production of fissile material.

There are some forms of power reactor that present additional opportunities for unreported fissile material production that must be addressed when designing a safeguards approach for the reactor.

Attention must be paid to multi-purpose small reactor designs that are principally designed for power production but also allow access to neutron beam ports for irradiation studies and isotope production. The Argentine designed CAREM reactor is an example of the multi-purpose small reactor – it has the potential to be an extremely valuable contribution to the nuclear industry – but its utility needs to be taken into account in the design of the safeguards systems applied to this new reactor type.

Special attention is paid to reactors that can be fuelled while on-line (OLRs) – these include some natural uranium fuelled graphite moderated reactors, pebble bed HTGRs and PHWRs. The capacity to move fuel through the core at a faster rate than has been declared opens a fissile material acquisition path that is not readily available to more conventional reactors – and the advantage of more favourable isotopic composition from lower burnups. The regular movements of spent fuel from the reactor also provide cover for the movement of undeclared material (e.g. by the production of a transfer flask with the same external appearance as a declared flask but with a greater capacity to allow for the removal of undeclared material).

While it is clear that that some reactor designs are especially suited to unreported production of fissile material (OLRs, multi-purpose reactors, reactors with declared dummy assemblies and any reactor with open structural areas within the reactor pressure vessel), there does not appear to be any practical reactor design in which it is possible to eliminate the possibility for unreported fissile material production entirely.

The scenario of unreported fissile material production is somewhat less complicated in the case of reactors, which only allow access to the core during refuelling. The use of containment and surveillance measures can allow the IAEA to derive a credible assurance that there has been no opportunity to remove unreported fissile material from the facility. Therefore, when the inventory of spent fuel at the facility is verified, the IAEA can indirectly derive assurance that there has been no unreported production of fissile material.

As there are inherent difficulties involved in any attempt to "prove a negative", the IAEA has always found the unreported production of fissile material to be a difficult scenario to cover effectively at a number of facilities. Relatively minor problems have the potential to prevent the IAEA from being able to derive an independent assurance that there has been no such unreported production of fissile material at a given facility. Any steps taken at the design phase of the reactor to limit the opportunity to misuse a reactor in this way will have substantial benefits for the IAEA and, in the long run, for the operator.

4. REDUCING THE STRATEGIC VALUE OF MATERIAL

As mentioned earlier, we see at least three basic approaches to enhance proliferation resistance of small and medium-sized reactors, namely: (1) by reduction of the strategic value of the materials involved in nuclear power generation; (2) by incorporating reactor design features preventing diversion of material; and (3) facilitating safeguards implementation.

In general, any reduction in the strategic value of material will simplify the task of the design of a safeguards system for the facility and make safeguards less intrusive for the reactor operators.

Conceptually there are a number of ways in which the strategic value of the material can be controlled:

- reduce the concentration of the fissile material (thereby increasing the quantity of spent fuel that must be diverted to obtain a significant quantity of IDU);
- increase the chemical barriers to the diversion of the material (producing fuel of a form that has features that present difficulties for reprocessing and recovery); and
- reduce the isotopic quality of the material (introduce features into the fuel that ensure that the final isotopic composition of the irradiated material is unsuitable for weapons purposes).

4.1. Reducing Concentration

Most power reactors are considered by the IAEA to be *item* facilities. This means that when the IAEA is designing the safeguards approach for the facility it considers that the fuel assemblies are to be accounted for as discrete, identifiable, individual items. Spent fuel items that contain less (preferably much less) than one *significant quantity* (SQ) of IDU [4] are subject to less intrusive safeguards than items that contain more than one SQ. In general safeguards on a large number of items with a low fissile material content will be less intrusive and simpler than safeguards on a small number of items with a high fissile material content. For example – CANDU fuel bundles contain very little IDU per assembly and, once discharged, are subject to only limited safeguards (the major complication arising from the safeguarding of CANDU reactors relates to the fact that fuel can be discharged while the reactor is operating).

4.2. Increasing the Chemical Barrier

If the fuel at a facility has features that render it unsuitable for reprocessing and fissile material recovery there is a case to be made for substantially decreasing the intrusiveness of the safeguards applied to the facility as part of the application of an Integrated Safeguards regime.

Silicide (and to a lesser extent carbide) fuels present substantial difficulties for existing reprocessing technologies when compared with oxide or metal fuels. The material is not completely intractable, but the processing of this material to recover fissile material is substantially more difficult than for most other fuel forms and, in general, it would require far longer conversion times to produce useable weapons components.

Under an integrated safeguards system the longer conversion times required for fuels which cannot readily be reprocessed can be taken into consideration in determining the inspection frequency and the intrusiveness of the inspection measures applied to the facility. It should be noted that choosing an intractable fuel form might have substantial fuel management implications and it would have to be considered in the context of an overall fuel cycle strategy.

4.3. Reducing the Isotopic Quality of the Material

Currently safeguards give only a limited recognition of the importance of the isotopic composition of the material to its proliferation significance. In the case of plutonium, for example, the only isotopic distinction that the IAEA currently acknowledges relates to the proportion of Pu-238 within a given batch of plutonium. Plutonium comprising 80% or more Pu-238 is acknowledged as being unsuitable for explosive use. For uranium the Agency recognises that uranium that is less than 20% enriched is of less immediate use to a proliferator than uranium enriched to 20% or greater.

As the safeguards system develops, there may be scope for recognising further distinctions in the isotopic composition of nuclear material. For example, if the material in question would require extensive processing facilities it will clearly be less desirable for a proliferator than material that is more readily applicable for weapons use and there may be scope for some reduction in inspection effort.

This line of reasoning can also be applied to the production of fuel for new reactor designs. As one example, if a particular proportion of Pu-238 degrades the utility of plutonium for explosive use, then introduction of appropriate (possibly quite small) quantities of Pu-238 at the fabrication stage may render the resulting spent fuel unattractive to potential proliferators. While the "spiking" of fuel would complicate the storage and handling of fresh fuel and have some effect upon the reactivity of the reactor these costs may be acceptable if they result in spent fuel that has a high intrinsic proliferation resistance. It may be possible to reduce the safeguards applied to such material to a much lower level than would otherwise be possible.

5. DESIGN FEATURES PREVENTING DIVERSION OF MATERIAL

5.1. Radiation Field

The radiation hazard associated with nuclear material is a substantial proliferation barrier due to the external dose potential to humans and the damage the radiation field could inflict on the equipment and non-nuclear materials needed to manufacture a complete operational nuclear device. The effectiveness of radiological barriers could be characterised by the associated dose rates or the time required for the accumulation of the lethal dose.

Thus materials could be categorised by the degree of remote handling required: starting with those suitable for unlimited hands-on handling and ending up with materials requiring fully remote and/or shielded facilities.

5.2. Facility Unattractiveness

The extent to which civil nuclear fuel cycle facilities are resistant to modifications required to convert them to the production of weapons-useable materials is another important intrinsic proliferation barrier. Those facilities, equipment and processes that cannot be modified to produce weapons-useable material would have a higher proliferation barrier.

A number of attributes can be used to characterise facilities by this criterion:

- the complexity of modifications needed to convert the facility to production of weapon-useable materials, including the need for additional specialised equipment, materials and technical knowledge;
- the availability of such specialised skills, material and knowledge to the country of proliferation concern;
- the safety implications of the facility's modification;
- the time and effort required to perform such modifications;
- facility throughput or, in the case of reactors, power level;
- environmental signatures associated with facility modification and misuse.

5.3. Access to Material

The extent to which facilities and equipment inherently restrict access to fissile materials represents an important barrier independent from institutional barrier including security and access controls that limit access.

Limiting the lifting capacity of cranes in the pond area and designing the structural limitations of the reactor area to ensure that there are only a limited number of possible paths for spent fuel to follow can serve as a useful adjunct to other proliferation limitation strategies.

6. DESIGN FEATURES FACILITATING SAFEGUARDS IMPLEMENTATION AT REACTORS

Safeguards are most easily applied to facilities in which movements of fuel and all other general maintenance activities are conducted exclusively during refuelling outages. Any equipment hatches must be able to be readily sealed and remain sealed for the entire time between refuelling outages. Provision of suitable locations for the attachment of seals should be incorporated into hatch design. Personnel hatches should be designed so that it is impossible for them to be used as an exit point for fresh or spent fuel.

If spent fuel is to remain on the reactor site between refuelling operations, it should be stored either in spent fuel ponds inside the reactor containment building, or transferred to separate storage ponds outside the reactor containment by a transfer channel, designed so that it can be readily sealed between refuellings. Provision of suitable locations for the attachment of seals should be incorporated in the design of the transfer channel – many existing facilities are difficult to safeguard satisfactorily because the transfer channel cannot be sealed effectively.
If spent fuel is stored outside of the reactor containment, the engineering design of the transfer channel should be such that the only possible path for spent fuel is between the reactor and the storage ponds. The external storage pond area should be designed so that the only time its cask transfer hatches need to be unsealed is when an offsite transfer of spent fuel is taking place. Additional "safeguards-friendly" engineering measures include ensuring that cask transfer hatches can only be opened if the transfer channel from the reactor containment has been closed and sealed (this ensures that there is no path for the removal of unreported fissile material from the core).

During refuelling operations, the IAEA generally maintains continuity of knowledge on the material in the core and covers the "unreported production" scenario by the use of surveillance systems. Provision of suitable places for the mounting of cameras and placement of recording equipment should be included in the design of the reactor hall.

7. CHOOSING THE BEST NUCLEAR FUEL CYCLE

7.1. Basic Criteria

There are at least three basic criteria, which are primary considerations in the selection of the future reactor system and associated nuclear fuel cycle:

- strategic considerations such as the State's independence of external energy suppliers, technological capabilities;
- economics, involving all costs, not just the cost of generating electricity, but the consideration of financial risks that could affect the investment as well;
- public acceptance factors incorporating safety, environmental considerations, and proliferation-resistance.

As US experts (TOPS) [5] have pointed out, economics will, by far, be the principal consideration in future decisions to build new nuclear plants. Considerations related to public acceptance would probably be secondary to, and influenced by, those related to economics. Commercial plant buyers are unlikely to view proliferation resistance as a high priority, relative to economic factors.

For the large capacity nuclear generating plants that have been favoured throughout the developed world, the capital costs of building plants and their associated infra-structure have tended to dominate the decision making process. The input cost of fuel has been a relatively small component of running costs of a plant, the capital cost tends to dominate all considerations. As these are major capital works it becomes difficult for any concern, beyond immediate economics, to influence design considerations – delay and expense are seen as impossible barriers to changes in plants' designs.

Plans for smaller more modular designs, with emphasis on distributed production and responsiveness to end-consumer needs, could drastically change these considerations as time goes forward. Physically small units, with small power outputs and lower overall costs (though not necessarily cheaper on a per kilowatt basis) could dominate the future deployment of nuclear power plants. As noted earlier, the costs associated with long distance electricity transmission and attendant transmission infrastructure tend to limit the per kilowatt advantage that large centralised plants have over smaller plants in the vicinity of demand centres.

With smaller capital costs and shorter deployment cycles, the concentration of risk is less significant and the chance for concepts of proliferation resistance to influence the overall design may become greater.

CONCLUSIONS

Developments in the nuclear industry and in nuclear technology should be considered in the context that the overwhelming majority of countries have given political and legal commitments against the acquisition of nuclear weapons. These commitments are reinforced by the institutional arrangements of the non-proliferation regime, especially by IAEA safeguards, and also by limits on the supply of sensitive technology. Institutional aspects of the non-proliferation regime continue to evolve, e.g. through strengthened safeguards, enhanced transparency and current progress towards Integrated Safeguards regimes as more States bring the Additional Protocol into effect.

Consideration of safeguards issues at the design stage of small power reactors can greatly benefit the safeguards that are applied by the IAEA to the facility. In an appropriately designed nuclear facility, a simple system of unobtrusive safeguards should provide confidence to the international community that the facility does not represent a risk of proliferation.

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FEASIBLE ADVANCED FUEL CYCLE OPTIONS FOR CANDU REACTORS IN THE REPUBLIC OF KOREA

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Abstract

Taking into account the view points on nuclear safety, nuclear waste, non-proliferation and economics from the public, international environment, and utilities, the SEU/RU and DUPIC fuel cycles would be feasible options of advanced fuel cycles for CANDU-PHWRs in the Republic of Korea in the mid- and long-terms, respectively. Comparing with NU fuel, 0.9 % or 1.2 % SEU fuel would increase fuel burnup and hence reduce the spent fuel arisings by a factor of 2 or 3, and also could reduce CANDU fuel cycle costs by 20 to 30%. RU offers similar benefits as 0.9% SEU and is very attractive due to the significantly improved fuel cycle economics, substantially increased burnups, large reduction in fuel requirements as well as in spent fuel arisings. For RU use in a CANDU reactor, re-enrichment is not required. There are 25,000 tes RU produced from reprocessing operations in Europe and Japan, which would theoretically provide sufficient fuel for 500 CANDU 6 reactor-years of operation. According to the physics, thermal-hydraulic and thermal-mechanical assessments of CANFLEX-0.9% RU fuel for a CANDU-6 reactor, the fuel could be introduced into the reactor in a straight-forward fashion. A series of assessments of CANFLEX-DUPIC physics on the compatibility of the fuel design in the existing CANDU 6 reactors has shown that the poisoning of the central element of DUPIC with, for example, natural dysprosium, reduces the void reactivity of the fuel, and that a 2 bundle shift refuelling scheme would be the most appropriate in-core fuel management scheme for a CANDU –6 reactor. The average discharge burnup is ~15 MWd/kgHE. Although these results have shown promising results for the DUPIC fuel cycle, more in-depth studies are required in the areas of ROP system, large LOCA safety analyses, and so on. The recycling fuel cycles of RU and DUPIC for CANDU are expected to achieve the environmental 3R's (Reduce, Reuse, Recycle) as applied to global energy use in the short- and long-terms, respectively.

1. INTRODUCTION

The Republic of Korea is a unique country in the world, having both PWRs (12 units) and PHWRs (4 units) in operation with an installed generation capacity of 13,716 MWe, which accounts for about 27% of the domestic installed electric-generation capacity. The installed electric-generation capacity of the four CANDU-PHWRs is 2,779 MWe. The fuel types for the Korean PWR and CANDU power plants are ~3-4 % slightly enriched uranium (SEU) and natural uranium (NU) dioxide fuels, respectively. Therefore, the Republic of Korea can exploit the natural synergism between these two reactor types o minimize overall waste production, and to maximize energy derived from the fuel. The synergism can be exploited through several different fuel cycles such as Recovered Uranium (RU) from irradiated fuel, Mixed Oxide $(Pu,U)O_2$ fuel (MOX), the TANDEM fuel cycle, and DUPIC (Direct Use of PWR spent fuel in CANDU reactors). These fuel cycle options are very attractive with respect to the NU and fuel disposal requirements per unit of energy produced [1].

However, the choice of the fuel cycle options for CANDU reactors in the Republic of Korea should take account of the domestic and international environments concerning non-proliferation in the Peninsula of Korea [2]. Considering the potential for PWR/CANDU synergism, and the requirement for non-proliferation, a guideline to select advanced fuel cycle options for CANDU reactors in the Republic of Korea could be as follows:

- The fuel cycle shall enhance reactor and fuel safety and operating margin.
- The fuel cycle has no involvement and no use of enrichment and reprocessing technologies in the Republic of Korea.
- The fuel cycle shall be compatible with existing reactors without major change of hardware.
- The fuel cycle shall improve the economics by means of reduction of fuel cycle and/or reactor operating costs.
- The fuel cycle shall be proliferation resistant.

According to these guidelines, the fuel cycles for CANDU reactors in the Republic of Korea would be NU, SEU, RU, DUPIC, and once-through-thorium (OTT) cycles. This paper will focus on the SEU, RU, and DUPIC fuel cycle options.

2. TECHNICAL FEASIBILITY OF USE OF 0.9%-EQUIVALENT SEU/RU IN CANDU

0.9 % or 1,2 % SEU fuel would increase fuel burnup and hence reduce the quantity of spent fuel produced by a factor of 2 or 3 compared with NU fuel[1]. The SEU fuel would reduce uranium requirements per unit energy by about 24 % and so improve uranium utilization, and would also reduce CANDU fuel cycle costs by 20 to 30% compared with NU fuel. RU offers similar characteristics and benefits as 0.9% SEU. The total amount of RU produced from reprocessing operations in Europe and Japan is around 25,000 tes with additional quantities from reprocessing in the former Soviet Union [3]. It is anticipated that RU can be obtained at very attractive price, because some utilities pay for the storage of the RU. Security of supply is not an issue, as SEU of equivalent enrichment can be substituted. The technical feasibility of using RU as a fuel cycle option for CANDU reactors in the Republic of Korea will be discussed as follows.

2.1. Carrier for CANDU Advanced Fuel Cycles

Since the early 1990’s, Korea Atomic Energy Research Institute (KAERI) and Atomic Energy of Canada Limited (AECL) have pursued a collaborative program to develop, verify, and prove the CANFLEX (CANDU Flexible fuelling) new 43-element fuel bundle design.. The CANFLEX [4] fuel bundle enables the introduction of advanced fuel cycles such as SEU, RU and other fuel cycles into CANDU reactors. The bundle assembly and its critical-heat-flux (CHF) appendages offer higher operating and safety margins than current fuel and the potential of reactor power uprating, which would further increase the economic competitiveness of the CANDU reactor, while maintaining full compatibility with operating CANDU reactors. It enables a higher power to be realized before CHF occurs, leading to a net gain in critical channel power (CCP) typically of 6 to 8% over the existing 37-element NU fuel. The greater element subdivision and the use of two element sizes lower the peak linear-element rating. Therefore, it is well suited for use in advanced fuel cycles, particularly those that can attain high fuel burnup. The fuel has been verified through extensive testing by KAERI and AECL and has been critically reviewed under a Formal Design Review. The compatibility of the fuel type with existing reactor systems has been proven through a demonstration irradiation of 24 CANFLEX-NU bundles in the Pt. Lepreau Generating Station at New Brunswick, Canada between September 1988 and August 2000 [5].

2.2. Physical Properties of RU

RU produced in the reprocessing facilities is stored in the form of slightly enriched UO_3 or U_3O_8 powder that has been converted from UNL (Uranyl Nitrate Liquor), with the majority of the material at an enrichment level up to 1% ^{235}U. RU is composed of ^{232}U, ^{234}U, ^{235}U, ^{236}U and ^{238}U isotopes and some of their daughter products. Traces of transuranic elements such as Pu, ^{237}Np, ^{242}Am and fission products such as ^{90}Sr and ^{106}Ru remain in RU. The level of ^{235}U enrichment in RU depends on both the type of reactor from which it came, and the adopted core management strategy. For example, a spent fuel of 33,000 MWd/MTU burnup in a 900 MWe PWR with fresh fuel of 3.25% ^{235}U contains 0.92%wt ^{235}U following 290 full power equivalent days in a 1/3-core refueling scheme. The main determinant in CANDU reactor physics with RU is the ^{235}U level. The ^{232}U assay is closely connected to the initial ^{235}U assay and is very low, and as well, its neutron absorption cross section is very low. The influence of the ^{232}U isotope upon reactivity in the CANDU 6 core is therefore negligible. Also, the influences of ^{234}U and ^{236}U upon reactivity in the CANDU 6 core are negligible, even if those isotopes are neutron absorbers.

RU contains typically about 1 ppb ^{232}U that decays with a half-life of 69.8 years. The daughters in the ^{232}U decay chain are removed during reprocessing but grow during storage. The first daughter in the chain is ^{228}Th with a half-life of 1.9 years. Since all other daughters in the chain have much shorter half-lives, including the radiologically important ^{208}Tl and ^{212}Bi, they are all in secular equilibrium with ^{228}Th. Therefore, the ^{228}Th build-up governs the rate of build-up of gamma activity and indicates the gamma activity with time relative to the quasi-equilibrium level attained after about 10 years. RU contains ^{234}U that contributes to a high specific alpha activity compared to NU. However the level is about the same as in conventional enriched PWR fuel, since the source of the increased ^{234}U is the initial enrichment of NU. RU also contains trace fission product gamma and beta emitters, and transuranic alpha emitters. Their quantities are negligible, but they contribute to total radioactivity of RU. This compels an RU processor to take proper protection against radiation hazards. Since ^{232}U daughter products develop very fast, their presence and related radioactivity cannot be avoided during the initial stages of RU recycling.

2.3. Reactor Physics of CANFLEX-RU in CANDU 6

Fuel management for CANFLEX-0.9% SEU and -0.9% RU has been simulated using 2-bundle-shift and 4-bundle-shift bi-direction fuelling schemes [6]. The results confirm that the CANFLEX-0.9% SEU or -0.9% RU fuel bundles fueled using a two- or four-bundle-shift refueling scheme would meet CANDU fuel performance criteria.

In KAERI,. the time-average characteristics of an equilibrium CANDU 6 core, and a 1200 full power day (FPD) refueling simulation using CANFLEX-0.92 % RU with 4-, 2- and 3-bundle refueling schemes have been simulated using the reactor physics computer code suite of WIMS-AECL[7]/DRAGON[8]/RFSP[9]. The isotopic composition of the RU dioxide pellets is assumed to be ^{232}U of 0.575 ppb, ^{234}U of 0.016%wt, ^{235}U of 0.90%wt, ^{236}U of 0.34%wt, and ^{238}U of 98.74%wt. The 0.92 % RU is equivalent to 0.90 % SEU. As shown in Table 1, the 4-bundle refueling scheme shows that the maximum channel power and maximum channel power peaking factor are too high to maintain the available operating margin, but the 2-bundle refueling scheme shows that a sufficient operating margin can be secured. Considering these operating parameters and refueling time, it could be recognized that the 3-bundle refueling scheme is preferable in comparison to the 4- or 2-bundle refueling schemes.

TABLE I. CANFLEX-RU MAJOR OPERATING PARAMETERS FROM CANDU-6 1200FPD REFUELLING SIMULATION

	CANFLEX-RU (4 BS*)	CANFLEX-RU (2 BS*)	CANFLEX-RU (3 BS*)	Remarks(8 BS*) (CANFLEX-NU)
Max. Channel Power (kW)(average) * Licensing limit: 7300 kW	7,228 [6,982]	6,889 [6,742]	7,012 [6,844]	6,840 [6,745]
Max. Bundle Power (kW)[average] * Licensing limit: 935 kW	873 [824]	805 [786]	850 [798]	862 [828]
Max. Channel Power Peaking Factor	1.175	1.094	1.135	1.114
Ave. Refuelling Rate (channel/day)	2.16	4.32	2.88	2.16
Ave. Refuelling Time (minutes/day	185.0	276.5	223.4	278.5

*BS = Bundle Shift in the refueling scheme.

Because of the neutron efficiency of CANDU reactors and the neutronic characteristics of RU, the RU with 0.92% wt ^{235}U can be burned as-is in CANDU reactors, without re-enrichment, to obtain about twice the burnup of NU fuel. It is noted that the reuse of RU in PWRs or BWRs will require re-enrichment. The ^{235}U would be burned down to low levels (i.e., 0.2 to 0.3%) in CANDU reactors compared to PWRs (0.8 to 1%). Both the annual throughput of RU fuel bundles into a CANDU 6 core and the annual spent fuel rate are 44.2 U tes a year. This quantity is about 50% smaller than that of NU fuel bundles. Therefore, the 25,000 tes RU produced in Europe and Japan as mentioned before would theoretically provide sufficient fuel for 500 CANDU 6 reactor years operation, since the initial core load of uranium for a CANDU 6 reactor is 85 tes, and the annual refueling requirement for an RU fuel burnup of 13 MWd/kgU is around 50 t/a. Although much of the RU to arise from reprocessing will be owned by utilities that may recycle the material in their own reactors, there is no doubt that sufficient RU will be available post 2000 to fuel CANDUs world-wide.

The extra fissile content of RU compared to NU offers greater flexibility in reactor and bundle design. In new reactor designs, or in existing reactors where there is sufficient heat removal capacity, RU offers a power uprating capability instead of gaining increased burnup benefits, by flattening the channel radial power distribution across the reactor core. This option involves trading-off the extra burnup potential of RU (greater neutron leakage from the core) against more power output. Using power flattening to obtain more power from a given-sized core has advantages in lowering capital costs relative to simply adding more channels to the reactor.

2.4. Thermalhydraulic and Thermal-mechanical Performance of CANFLEX-RU in CANDU 6

The thermalhydraulic design characteristics of RU fuel bundles in a CANDU 6 reactor have been studied [10] by comparing channel axial heat flux distributions (CAFDs) and bundle-radial heat-flux distributions (BRFDs) of CANFLEX-NU and –RU bundles in a CANDU 6 reactor, and then by evaluating the CCP with the fuels. The CAFD profile of RU fuel is flat and slightly concave in the channel center region, because of the 4-, 2- or 3-bundle shift refueling scheme, compared to the AFD of those channels fuelled with NU fuel by an 8 bundle shift fuelling scheme. Considering both CAFD and BRFD, the RU fuel increases the CCP by about 1%, compared with that of the CANFLEX NU fuel. Therefore, the thermalhydraulic performance of the RU fuel bundle will maintain the merits of the CANFLEX NU fuel bundle, such as the enhancement of thermal margin, compared with the existing 37-element NU fuel.

Using the power envelopes based on the 4-bundle shift refueling simulation results, RU fuel element performances were evaluated with ELESTRES code [11]. It was showed that RU fuel with extended burnup will show good in-reactor performance as existing CANDU fuels do, because the fuel temperature and element internal gas pressure are far below the design criteria. The total hoop strains of the sheath are within reasonable range in terms of sheath plastic strain. The sheath strain and fission gas release predicted at end-of-life are compatible with those found from 20 years experience with CANDU fuel [12].

2.5. Fabrication and Handling of CANFLEX-RU Fuel

Three main aspects differentiate RU fuel fabrication from NU or SEU fuel fabrication: higher specific activity of the material, criticality considerations and the increase in specific gamma activity related to the in-growth of ^{232}U decay daughter products. Use in CANDU does not require re-enrichment of the ^{235}U in RU, and so there is no re-enrichment of ^{232}U, ^{234}U and ^{236}U. Thus the radiological implications of handling RU for CANDU use are greatly reduced, compared with those where re-enrichment of RU is required for, for example, PWRs. An assessment of the health physics aspects of manufacturing and handling RU fuel for CANDU reactors was done in a joint program between BNFL (British Nuclear Fuel plc.), KAERI and AECL. BNFL converted reprocessed spent PWR fuel into 200 kg of UO_2. The conversion took place one year after reprocessing. The characteristics of the recovered UO_2 powder met CANDU specifications in terms of both chemical impurity contents and physical characteristics. The powder was granulated and pressed into green pellets which were sintered under the normal conditions for CANDU fuel. During the sintering, the release of ^{137}Cs and other volatile fission products from RU was below detectable levels. The finished pellets met all the physical and chemical specifications for CANDU fuel. Activity level measured on the finished RU bundle was 1.3 times higher than those of NU bundles, when measured at 30 cm distance.

All aspects of the transport of RU powder, for example, within the United Kingdom (UK), to the Republic of Korea and within the Republic of Korea have been examined jointly by BNFL and KAERI in relation to both countries' regulations and international regulations as set out in IAEA (International Atomic Energy Agency) Safety Standard Document – Regulations for the Safe Transport of Radioactive Material, 1996 Edition NO. ST-1. Also considered is the transport of SEU derived from NU as an available substitute for RU fuel. In the spring of 1998, a transport of BNFL recovered-UO_2 and slightly enriched UO_2 powders from BNFL in Springfield, UK to KAERI in Taejon, Republic of Korea was successfully carried out as part of the KAERI/BNFL development program collaboration on the RU fuel for CANDU reactors. This experience demonstrated that there are no difficulties involved in the transport of RU as either non-fissile material (<1.0% ^{235}U) or fissile material ($\geq$ 1.0% ^{235}U) as defined in the above IAEA document. No obstacles to the transport of commercial quantities of recovered UO_2 powder from the UK to the Republic of Korea are also recognized.

2.6. Fuel Cycle Costs for RU

Most countries and/or utilities, which adopt a reprocessing strategy, do so for strategic energy self-reliance and/or for waste management reasons. Generally, RU is owned by the utility that contracts for reprocessing of spent fuel. The uranium and plutonium recovered from reprocessing are often held as "low or zero cost" stocks by the utilities. Hence, there is the possibility that RU will be competitively available on the open market. The potential annual saving to a CANDU utility by the utilization of RU is significant, but strongly dependent on the price paid for the RU powder and fuel fabrication.

KAERI [13] estimated relative annual savings of RU fuel relative to existing 37-element NU fuel in a CANDU 6 reactor by assuming that the fabrication cost of the RU fuel bundle is about 16% higher than that of the existing fuel. For example, with recycled UO_2 priced at 100% of the natural UO_2 cost and conversion cost of US$12/kg, the annual fuelling cost of the RU fuel bundles would show a saving of 22% relative to that of the existing fuel. Ongoing work will reduce the uncertainties in the fuelling costs for RU, namely the cost of ceramic-grade UO_2 powder, and the cost of RU fuel fabrication.

AECL conducted a cost assessment of the impact of SEU on spent CANDU fuel disposal cost [14]. SEU enrichments equivalent to that of RU could result in a 20% reduction in annual spent fuel disposal costs compared to NU fuel. So, the use of RU in CANDU reactors would appear to be an extremely attractive way of dealing with a waste product while at the same time extracting additional energy.

3. DUPIC FUEL CYCLE

Considering that spent PWR fuel contains enough fissile materials to be burned in CANDU reactors, DUPIC [15] involves converting the spent PWR fuel into CANDU fuel by a thermal-mechanical dry process without any wet chemical processing. DUPIC fuel cycle technology is currently under development by KAERI and AECL in cooperation with the US Department of State and the IAEA.

3.1. Status of DUPIC Fuel Cycle Development in the Republic of Korea

The DUPIC fuel cycle is being developed in a phased approach. Phase I was a feasibility study, which was conducted between 1991 and 1993, to conceptually evaluate several possible DUPIC fuel fabrication processes. Among several fabrication options, the OREOX (Oxidation/Reduction of Oxide fuel) process was selected as the optimum DUPIC fuel fabrication method for further study [16], and its safeguardability was judged to be achievable. Phase II, which is currently under way from 1994 to 2002, is focused on demonstrating that DUPIC fuel can be fabricated using the OREOX process on laboratory scale, and on assessing the fuel performance by irradiating the fuel in research reactors. Both KAERI and AECL are developing the fuel fabrication technology and also assessing the reactor physics and the fuel element performance. KAERI is developing the safeguards methods through international collaboration with LANL (Los Alamos National Laboratories) in USA and with the IAEA.

Following a series of hot cell experiments, AECL has successfully fabricated three DUPIC fuel elements that have been undergoing irradiation in the NRU research reactor at AECL-CRL since 1999 March. In 1999, KAERI completed preparations for hot cell equipment, refurbishment of a hot cell for the fuel fabrication, and the verification of safeguards equipment. Using about 1 kg of spent PWR fuel, a characterization study on DUPIC powder and pellets has successfully been performed in a hot cell of PIEF (Post Irradiation Examination Facility) in KAERI. Subsequently, KAERI has fabricated three DUPIC fuel mini-elements with 50 pellets at the remote fuel fabrication laboratory in the IMEF (Irradiation Materials Examination Facility). The KAERI-made DUPIC elements have been irradiated for fuel performance evaluation in the HANARO research reactor since 2000 May.

3.2. Reference Composition of DUPIC Fuel

As spent PWR fuel composition depends on the initial enrichment and burnup conditions, the composition of DUPIC fuel is not uniquely defined. The fissile materials in the fuel contain nominally about 0.9%wt of ^{235}U and about 0.6%wt. of ^{239}Pu and ^{241}Pu. To reduce the effects of composition heterogeneity (fissile content and neutron absorbing fission products) on core performance, a reference DUPIC fuel composition (see Table 2) was determined to make maximum use of 36000 spent PWR assemblies currently in the Republic of Korea [17]. The enrichments of ^{235}U and ^{239}Pu in reference fuel are 1.0 and 0.45%wt, respectively., which were determined by that DUPIC core performance is comparable to that of a NU core with high utilization of spent PWR fuel and low cost of fuel cycle. The reference enrichment can be achieved by mixing two spent PWR fuel assemblies of higher and lower ^{239}Pu content and, if necessary, blending SEU and depleted uranium (DU). Under this condition, it is possible to utilize 90% of spent PWR fuels in the Republic of Korea as the DUPIC fuel formula. On average, the amounts of

SEU and DU used for blending would not exceed 8.6 and 10.6%, respectively, of the mass of candidate spent PWR fuels.

TABLE II. COMPOSITION OF DUPIC FUEL (KG)

Isotopes	^{234}U	^{235}U	^{236}U	^{238}U	^{237}Np	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{241}Am	^{241}Am
Weight*	0.00002	0.16291	0.08088	17.20170	0.00699	0.09766	0.04059	0.01479	0.00934	0.00944	0.00001	0.00193

3.3. Reactor Physics Compatibility of DUPIC Fuel with CANDU 6

A series of CANFLEX-DUPIC physics studies has been performed to assess the compatibility of the fuel design with the existing CANDU 6 reactors [18]. Since DUPIC fuel has a higher Pu content than usually found in CANDU fuel, poisoning of the fuel with, for example, natural dysprosium is suggested to provide a safety feature of the DUPIC core in a CANDU 6 reactor. According to a sensitivity study on the poisoned DUPIC fuel core performance in a CANDU 6 reactor, a 2-bundle shift refuelling scheme was shown to be the most appropriate option for the in-core fuel management scheme, which resulted in an average discharge burnup of 15 MWd/kgHE. Under this fuel management strategy, the CANDU 6 reactivity control devices of the light water zone controller unit, adjuster and mechanical control absorber would maintain their functional requirements for the DUPIC fuel system. Adjusting the quantity of neutron poison in the central element controls the void reactivity of the fuel. By using a slow-burning poison such as dysprosium, the void reactivity is controlled throughout the life of a bundle. The average peak channel and bundle power in a CANDU 6 reactor are 6722 kW and 775 kW, respectively, which correspond to 7.9 and 17.1% margins to the license limits for the CANDU 6 operation, respectively. The maximum and average channel power peaking factors (CPPFs), defined as the ratio of instantaneous channel power over the reference one, are 1.15 and 1.07, respectively, over 600 FPDs, compared to an average CPPF of 1.064 for the NU core. Also, the refuelling simulations have shown that the DUPIC bundles possess sufficient margin to the SCC (stress-corrosion-cracking) threshold of UO_2 fuel, a potential failure mechanism for Zr-clad UO_2 fuel.. Though these are promising results of the DUPIC fuel cycle, more in-depth studies are required for the following areas: ① Performance of reactivity devices in the core including the xenon override capability of the adjuster system and the margin of the shut-down systems. ② Regional overpower protection (ROP) system for the reactor controllability during normal and transient operations. ③ Safety analysis for the large-break LOCA under various operating conditions. ④ Effect of the fuel composition on the core performances (fuel composition heterogeneity effect). ⑤ Performance behaviour of the fuel which contains initially solid fission products as impurities. ⑥ Economic assessment of the fuel cycle.

3.4. Possible Refuelling Routes for DUPIC Fuel into a CANDU 6 Core

For the refuelling of highly radioactive fresh DUPIC fuel, two refueling routes could be considered. First, the fresh DUPIC fuel could be loaded by a wet route, which follows the reverse path of the current CANDU 6 fuel discharge processes. That is, once the fuel is shipped from the fuel manufacturer to the spent fuel bay, it would be moved into the discharge bay inside the reactor building. After loading the DUPIC bundles in the discharge ladle, the fuel bundles would be elevated to the discharge port. New equipment would be needed to push the fuel bundles from the ladle into the fuelling machine magazine. Once the fuel is inserted into the fuelling machine, the remote operation mechanism of the current fuelling machine would allow refueling. The wet route is proposed to minimize the modification of current refueling system and radiation exposure to personnel. As an alternative, a dry route could be considered, which uses the same sequence as the current refueling system, but requires a new heavily shielded area and fully-automated remote operation. However, the optimum refueling system for commercialization of DUPIC fuel needs further investigation.

SUMMARY AND CONCLUSIONS

the Republic of Korea can exploit the natural synergism between the PWR and CANDU reactor types to minimize overall waste production, and to maximize energy derived from the fuel. The synergism can be exploited through a few different fuel cycles such as SEU, RU, and DUPIC fuel cycles, taking account of the international environments concerning non-proliferation in the Peninsula of Korea.

Comparing with NU fuel, 0.9 % or 1.2 % SEU would increase fuel burnup and hence reduce the quantity of spent fuel produced by a factor of 2 or 3, and also reduce CANDU fuel cycle costs by 20 to 30 %. RU offers similar benefits as 0.9 % SEU.

RU is one of the products of conventional chemical reprocessing of spent nuclear fuel. It is composed of uranium isotopes and some of their daughter products, traces of transuranic elements and fission products. The daughter products of uranium and traces of the transuranic elements and fission products are in negligible minute quantities and will not influence the CANDU core, fuel performances, manufacturing, and handling of the fuel. There are 25000 tes RU produced from reprocessing operations in Europe and Japan, which would theoretically provide sufficient fuel for about 500 CANDU 6 reactor years operation.

According to the physics, thermal-hydraulic and thermal-mechanical assessments of CANFLEX-0.92 % RU fuel for a CANDU 6 reactor, it was found that : ① The RU fuel can be burned as-is in CANDU 6 reactors, without re-enrichment, to obtain about twice the burnup of NU fuel, and so increase resource utilization ② The annual throughput of RU in CANFLEX bundles into CANDU 6 reactor core and the annual spent fuel are 45 U tons a year, which is reduced by about 50%, compared to that of the NU fuel. The lower volumes of fuel throughput and spent fuel have a positive effect on economic, environmental and public acceptance aspects of the fuel cycle. ③ The extra fissile content of RU compared to NU offers greater flexibility in reactor and bundle design. ④ It is anticipated that RU can be obtained at a very attractive price, because some utilities pay for the storage of the RU. Security of supply is not an issue, as SEU of equivalent enrichment can be substituted.

Increased concerns regarding the political and economical repercussions of spent fuel reprocessing and spent fuel disposal provide additional incentive to develop the advanced, innovative fuel cycle of DUPIC. Including the additional energy extracted from the fuel in a CANDU reactor, the potential benefits of the DUPIC fuel cycle in comparison with conventional wet reprocessing and with respect to uranium utilization and spent fuel arising are "proliferation resistance due to the non-separation of uranium, plutonium and fission products during the fabrication process", "a small amount of radioactive waste due to the nature of dry processing", "saving of uranium resources due to the efficient uranium utilization-the DUPIC fuel cycle could reduce the NU requirements by about 25 % compare with the direct disposal fuel cycle", and "a significant (a three-fold) reduction in spent fuel arising, compared with direct disposal fuel cycle". A series of reactor physics assessments have been made of the compatibility of the CANFLEX-DUPIC fuel design with existing CANDU-6 reactors for the reference enrichments of 1.0 w/o ^{235}U and 0.45 w/o ^{239}Pu in DUPIC fuel. It was found that: ① The poisoning of the central element of DUPIC fuel with, for example, natural dysprosium, reduces the void reactivity of the fuel. ② The 2-bundle shift refuelling scheme would be the most appropriate fuel management option for the CANDU 6, and results in an average discharge burnup of 15 MWd/kgHE. With the DUPIC refuelling scheme, the CANDU 6 reactivity control devices of the light water zone controller unit, adjuster and mechanical control absorber would maintain their functional requirements. ③ The DUPIC fuel bundles possess enough margin to the SCC threshold of UO_2 fuel. ④ Although the above results have shown promise, more in-depth studies are required in the following areas: "performance of reactivity", "ROP", "safety analysis for the large-break LOCA", "effect of the DUPIC fuel composition on the core performances", and "fuel performance behaviour".

In conclusion, the use of RU and DUPIC in CANDU has beneficial environmental impact on overall fuel cycles. The recycling fuel cycles of RU and DUPIC for CANDU are excellent examples of the environmental 3R's (Reduce, Reuse, Recycle) as applied to global nuclear energy use in the short- and long-terms, respectively.

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WATER COOLED REACTORS

(Session 5)

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UNATTENDED REACTOR FACILITY FOR AUTONOMOUS SMALL NUCLEAR POWER PLANT UNITHERM

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Abstract

The concept of small size co-generation nuclear power plant of enhanced safety intended for electricity supply to remote and difficult-to-access areas has been presented. The basic design features and the configuration of an integral nuclear reactor and plant as a whole are described.

1. THE CONCEPT OF SS NPP UNITHERM

Small size nuclear power plants (SS NPP) can become a reasonable option substituting the energy sources based on fossil fuel for remote and difficult-to-access regions. The application of SS NPP as the source of heat and power supply for industrial enterprises, local communities or regions may turn to be economically efficient and rather promising from the social and ecological viewpoints.

Unitherm SS NPP is intended to satisfy the energy demand primarily for the non-industrial sector in small communities with a population of 2 or 3 thousand people. The nominal capacity of the plant 15-30 MWt, has been chosen based on the requirements on mass and dimensions so as to make the separate units transportable. Typically, there are no large industrial enterprises in such communities such as are often found in the regions of Far North and East of the Russian Federation.

SS NPP can also be applicable for energy supply in towns and relatively large communities with a population of 10-15 thousand people. Such communities in regions having no central electricity transmission lines are few. They typically include any industrial enterprise such as a port, mining-processing plant, mine, any processing plant and also have certain infrastructure (social amenities, food processing plants, municipal and retailing facilities, etc.).

The concept of SS NPP design relies on some basic principles that determine the overall configuration, structure and characteristics of the plant equipment applied, manufacturing, assembling and operating technologies.

- SS NPP is intended mainly for remote-to-access regions with poorly developed infrastructure. Therefore the plant should be designed such as to permit its operation with minimum number of the operating personnel. Moreover, it is reasonable to exclude specialists having functional training directly related to nuclear reactor control from the enlisted personnel.
- Construction of such a plant in the distant region should not involve a large scope of erection and assembly works such as would be impossible without the development of a specialized construction service base, secondment of the construction team for a long period of time, etc. Summer temperatures are rather short in the regions of North and East of the Russian Federation, so the construction work can not be overextended.

- The plant should consist of a limited number of large factory-built assembly units transportable to the site. Prefabrication will guarantee quality and long service life of such units. Thus, erection works required on site will be minimized.
- To enhance safety and simplify the operating technologies, no refueling of the reactor core during its service life is envisaged. That means that the reactor core life should be estimated for 20-25 years with the capacity factor being 0.7-0.8.
- The plant should be capable of operating without any limits as to the number and rate of load-following regimes, thus the load could be rejected and restored without the need to shut reactor facility (RF) down.
- The concept of SS NPP development implies the use of the water-water reactor technologies well proven as propulsion reactors, though with the appropriate optimization of their characteristics.
- The design efforts should be focused on the application of the equipment found in the batch production and having the prototypes proven in operation.
- SS NPP of XXI century should ensure ultimately achievable safety based on such principles as the inherent safety, maximum number of barriers against release of radioactivity and the use of passive safety systems only.
- Upon expiration of the service life the plant should be dismounted and removed, while the site should be recultivated to "green field" status.

2. REACTOR FACILITY

The heart of Unitherm SS NPP, its source of energy, is the reactor facility (RF). Its flow sheet is schematically shown in Fig. 1. To guarantee that generated steam is not radioactive, a three-circuit system for energy transfer from the reactor core to the working fluid is provided in RF. In this case the loss of integrity in any of heat-exchange surface will not cause an accident situation.

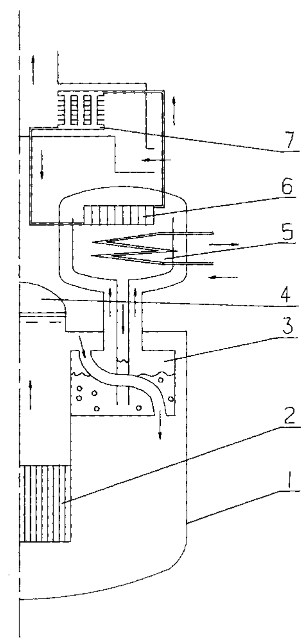

Fig. 1. Schematic diagram of the reactor facility: 1: reactor vessel; 2: reactor core; 3: heat exchanger; 4: pressurizer; 5: steam generator; 6: evaporator; 7: radiator.

The reactor core 2 placed in the reactor vessel 1 is cooled by the primary coolant driven by natural circulation and exposed to steam-gas environment in the pressurizer 4. In the intermediate heat exchanger 3 the energy released in the reactor core is transferred to the secondary coolant which goes upward to flow outside the tubes of the helical once-through steam generator 5. After cooling on the steam generator heat-exchange surfaces, the coolant is directed to the intermediate heat exchanger. The system consists of several parallel sections or units each designed as a thermal siphon placed in an individual casing. In case of a leak in one of the heat-exchange surfaces of the unit, the latter is isolated from the user by means of isolation valves installed in the tertiary circuit and without the need to shut the reactor facility down. A damaged unit can be replaced during the scheduled preventive maintenance.

The reactor core power varies depending on the steam load to the RF due to self-control as the temperature reactivity coefficient remains negative in the whole range of temperatures. Decrease of reactivity because of burnup and poisoning can be partly compensated by burnable absorbers and the temperature effect which ultimately amounts to 20°C/year. In turn, the latter can be compensated by motion of the reactivity control members during the periodic maintenance.

Further development of the RF layout was determined by the decision not to use the operating personnel for RF control. The thermal siphon unit is equipped with the evaporator heat exchanger being the part of the continuously operating fourth (independent) circuit for heat removal. Under normal operating conditions these capacity losses should be minimized. However, a sudden reduction or even a termination of heat extraction by the users should not cause a shutdown of the reactor and overshooting of the system parameters. This situation can be mitigated by the use of the independent circuit. In addition to the evaporator 6, the circuit consists of the radiator 7 connected to the evaporator and cooled by the atmospheric air under natural circulation (the fifth circuit). The independent circuit for heat removal allows the transfer of the reactor to hot standby mode without the need for its shutdown as a preliminary step. In accident situations the circuit acts as the emergency residual heat removal system.

Significant temperature changes and low negative temperatures in the winter season (from plus 35 to minus 55 °C) in the candidate sites of Unitherm SS NPP application predetermine the selection of the specific coolant in the independent circuit. The proposed options are, for example, ammonia or aqueous solution of ethylene glycol.

It should be noted that shutoff valves are not used at any RF pipelines, except for the user circuit, i.e. all systems are continuously in operation rather than just available. This feature contributes significantly to RF reliability and safety enhancement.

Fig. 2 shows the schematic diagram of RF layout. As can be seen, the facility is placed inside a reinforced concrete impact-proof containment. The reactor and its hydraulically connected systems are located in a strong leaktight safeguard vessel which completely confines the consequences of the design basis accidents involved a leak in the primary circuit. In addition, all equipment of the RF is located in the containment intended to reduce the radiation consequences of the beyond design basis (hypothetical) accidents.

3. LAYOUT OF THE USER CIRCUIT

The general scheme of SS NPP, its configuration and efficiency depend primarily on the functions assigned to the plant. Fig. 3 schematically shows the envisaged principal schematic solutions. In the general and simplest case (see Fig. 3a), the RF supplies steam to the main steam pipeline which feeds all available users such as turbine generator, heating boilers, etc. Exhaust steam and condensate are collected at a common point (for example, the condenser) and then water is directed to the steam generator inlet by the feed pump. The RF controller is able to maintain steam pressure in the steam pipeline by varying the flow rate of feed water. The controllers of user equipment can maintain preset parameters such as generator rpm, network water temperature, etc. using variations of extracted steam rate. A change in steam consumption by any user produces almost equivalent impact on steam capacity, i.e. power output of the RF.

The series and more economical scheme of the user connections is given in Fig. 3b. Typically, low capacity turbines have no interim extraction stages. In this case the whole amount of generated steam is directed to the electrical generator turbine operating under back-pressure. Heating boilers are installed at the turbine exhaust. The boiler controller produces a command for extraction of the required amount of steam. To provide for independent control of the turbine and boilers, there are pipelines bypassing the turbine to discharge exhausted steam to the condenser. Mutually related controllers installed at the bypassing lines maintain steam pressure at the exhaust. If the amount of steam extracted by the boilers is significant and back-pressure drops beyond the set value, the exhaust is made up by live steam, whereas in response to the rise of back-pressure in the case of an isolated bypass line, the exhausted steam is discharged to the condenser. Similarly to the previous case, steam pressure in the main steam pipeline is controlled by variations of feed water supply to the steam generator, and this relationship impacts RF power output by means of the temperature effect.

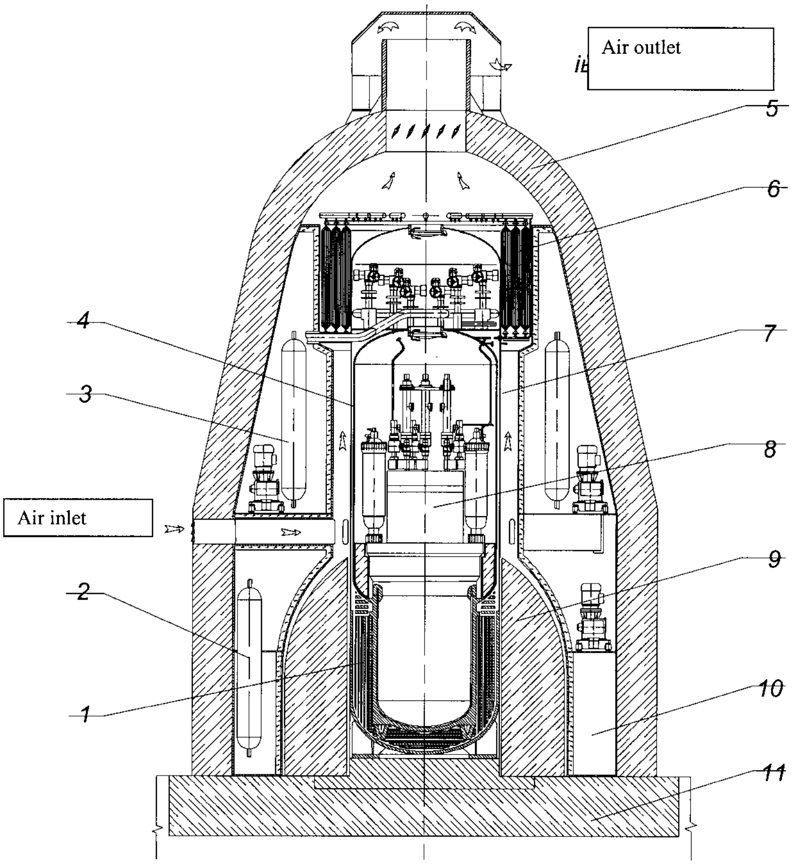

Fig. 2. Reactor facility: 1: iron-water shielding tank, 2: cylinders for storage of gaseous radwaste, 3: liquid absorber injection system , 4: containment, 5: impact-proof containment, 6: cooldown system heat exchanger, 7: safeguard vessel, 8: reactor, 9: biological shielding blocks, 10: storage for liquid and solid radwaste, 11: foundation.

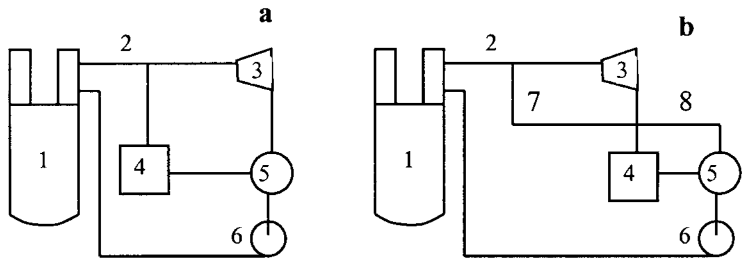

Fig.3. Schematic diagram of the thertiary circuit of SS NPP: parallel (a) and series (b) connection of the users. 1: RF, 2: main steam pipeline, 3: turbine generator, 4: boilers, 5: condenser, 6: feed water pump, 7: turbine bypass, 8: discharge line for exhaust steam.

A specific layout of SS NPP can be chosen proceeding from the configuration of the users and relations of their capacities.

Obviously, the three-circuit heat transfer scheme would cause a reduction of outlet steam parameters, though with enhanced safety. The thermal parameters of the coolants in RF circuits have been chosen from the well-proven range of working pressure and temperature values typical in the primary circuits of the water-water reactors. Account has also been taken of the experience in operation of propulsion nuclear power facilities under natural circulation of the primary coolant.

Based on the NPP design experience, the primary coolant temperature at the core outlet can be taken to be equal to 340°C. Considering the above-mentioned decrease of temperature at the end of life prior to the maintenance, this value would amount to 320°C. Under natural circulation of the primary coolant, the temperature difference at the core outlet and inlet will amount to no less than 70°C. Thus, the minimum temperature of the primary coolant will be 250°C.

Water is used as the coolant in the intermediate circuit. There are two options for the coolant flow: single-phase or two-phase fluid. The advantages and disadvantages of both options are obvious. Heat transfer with two-phase coolant is characterized by higher values of heat transfer coefficients, while due to latent heat of evaporation can be transferred by lower flow rate which is an important factor under natural circulation. Generally, this option is beneficial in terms of the space-saving characteristics of the design.

At the same time, two-phase thermal siphon poses limitations on maximum achievable temperature therein. In the ultimate case of using the intermediate heat exchanger evaporator, its temperature can not exceed 230°C (see Fig. 4a), and the coolant temperature in the tertiary circuit can not go above 210°C. The assumed minimum temperature head of 20°C is provisional, but in reality it helps to minimize the heat-transfer surface areas.

If the SS NPP is intended for heating purposes only, its potential is absolutely sufficient.

The combined generation of heat and electricity demands a certain efficiency and the associated conditions impact the choice of parameters for the tertiary coolant. It is desirable to draw on the experience in designing and operation of the steam turbine facilities that are considered as the operating prototypes. In particular, it would be possible to use the low capacity turbine generator units operating at low steam parameters. Such equipment is developed and produced by Kaluga turbine works.

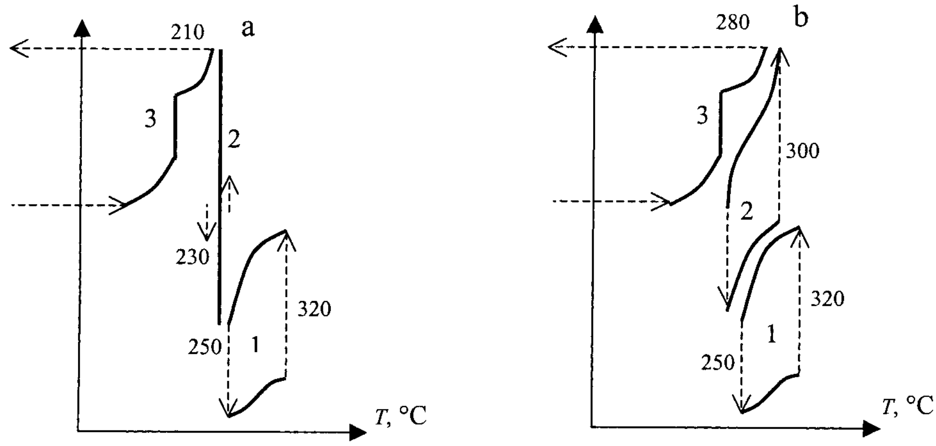

Fig. 4. Coolant temperature profiles in the primary, secondary and tertiary circuits of RF for two-phase (a) and single-phase (b) thermal siphones.

For example, the turbine generator TG2.5/6.3 equipped with the modified turbine R7/1.0 is able to generate 2.5 MWe in the condensing mode with consumption of 24 t/h steam at 1.2 MPa pressure and 210 °C temperature. The condenser pressure should be maintained at 0.02 MPa.

The use of modular geothermal power units for the container design, for example, the power unit "Touman-2" which is connected in series to heat generation unit GTS-700P, can produce 1.7-1.8 MWe and 20 MWt providing the following network water outlet parameters: 0.45 MPa pressure, 90 °C temperature, 715 t/h flow rate.

The use of single-phase convective thermal siphons will permit (see Fig.4b) an increase of the temperature of generated steam to 260 °C. It would be beneficial for the overall efficiency of the plant, though the RF dimensions have to be enlarged. It should be pointed out that in terms of cost efficiency, the increase of SS NPP efficiency can not be an end in itself, since the fuel cost is not critical for the price of electricity generated.

To ensure independence of the Unitherm SS NPP from the availability of water resources, it is proposed to use air-cooled condensers.

4. RELIABILITY AND SAFETY

High reliability and endurance of the RF can be guaranteed by the use of the materials and technologies of ship reactor design, which have been proven in operation for a long period of time. In particular, the reactor core can be fuelled with uranium-zirconium fuel elements with a metal-ceramic fuel matrix; tubes for the steam generators and heat exchangers can be made of titanium alloys. Shock-resistant structures will allow the system to withstand ultimate seismic loads. Water chemistry conditions are selected such as to drop continuously operating primary coolant purification systems and, as a result, the reactor design can be of integral layout which is advantageous to minimize the primary circuit boundaries.

There are no components in the RF that need to be moved during the plant operation. The circulation pumps, automatically operated control rods, valves and similar components are excluded. All changes in the operating modes will be initiated by natural processes.

The safety systems used in the RF are passive, they do not rely on external sources of power to implement their functions. In he case of initiating the emergency protection of the reactor core, the control members will be inserted in the core by gravity and by the energy of compressed springs. The continuously operating independent heat removal system will remove residual heat and cool the reactor facility down.

In the case of the primary circuit leak, the released steam-gas mixture will be confined within the safeguard vessel of the RF. As soon as the pressures in the reactor and the safeguard vessel are equal, the released flow will stop. In this situation the coolant level in the reactor will remain high enough that it will not hamper natural circulation in the primary circuit, removal of residual heat and RF cooldown.

The safety of the RF can be guaranteed by the inherent safety of the reactor core, its low energy rating and availability of five barriers against release of the radioactive products such as:

- metal fuel matrix of fuel elements,
- fuel cladding,
- primary circuit boundaries,
- safeguard vessel,
- RF containment.

Under normal operating conditions the level of ionizing radiation on the surfaces of the RF shielding structures will not exceed the natural background. In the event of the ultimate design basis accident the dose rate of ionizing radiation at a 100 m distance from the RF will exceed the background by just 10 %. Under all design and beyond design basis accidents the individual dose for the population will not exceed 0.11 rem per year.

The safety enhancement and non-proliferation of nuclear materials are the targets for choosing the long core life (20-25 years) which is equal to the service life of the reactor facility, thus eliminating the need for in-service refueling of the reactor core.

The Unitherm SS NPP has been designed to implement the full manufacturing, assembly and adjustment process in workshop conditions. Prefabricated large assembly units (15-20 pieces) with mass from 100 to 175 t will be delivered to the site where only a limited amount of mounting and pre-start activities should be performed. The assembly units can be transported by water-borne vehicles, large cargo platforms and tractor lorries. After decommissioning all components of SS NPP should be evacuated and the site can be renovated.

According to this concept, the maintenance and service of the RF of SS NPP would be provided by a regional center common for several plants and equipped with qualified personnel, advanced communication and transportation facilities.

IRIS: AN INTEGRATED INTERNATIONAL APPROACH TO DESIGN AND DEPLOY A NEW GENERATION REACTOR

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Abstract

IRIS (International Reactor Innovative and Secure) is a novel light water reactor, with a modular, integral configuration. A brief description of its design characteristics is provided along with a discussion how IRIS can accommodate interchangeable cores fueled by uranium oxide and MOX with varying fissile enrichment levels. The integrated international approach is presented by outlining the rationale and modalities behind the IRIS unique experiment at designing, conducting, deploying and servicing a nuclear energy system by an integrated international team of reactor vendors, component manufacturers, fuel vendors, architect engineers, utilities, laboratories and universities.

1. INTRODUCTION

IRIS (International Reactor Innovative and Secure) is a modular, integral, light water cooled, small to medium power (100-335 MWe/module) reactor which addresses the requirements defined by the US DOE for Generation IV nuclear energy systems, i.e. proliferation resistance, enhanced safety, improved economics and waste reduction. The technical characteristics of IRIS have been discussed in detail in several papers [1-6] and therefore will not be the repeated here at length However, a brief description will be given in Section 2 for the convenience of the reader. Rather, the focus of this paper will be on the integrated international approach which from the very onset has been the backbone of the IRIS concept and which is truly unique to IRIS A companion paper in this symposium [7] presents a fundamental, novel feature of IRIS, i.e. its “safety by design” approach, where accidents are “designed out” to the maximum extent possible, instead of engineering how to cope with their consequences

2. THE IRIS REACTOR DESIGN

IRIS features an integral vessel which houses the reactor core and support structures, core barrel, upper internals, control rod guides and drivelines, steam generators, pressurizer, heaters, internal spray located in the upper head and reactor coolant pumps (see Fig. 1) Such an arrangement eliminates a multi-loop configuration of steam generators and pressurizer, connecting pipes, and supports. Depending on the plant power rating, the vessel has a height of 18-22 m and an outside diameter of 4-6 m, a size which is within the state-of-the-art fabrication capabilities. The configuration shown in Fig. 1 is for a 300 MWt design. Hot coolant rising from the reactor core to the top of the vessel is pumped into the steam generator annulus by six reactor coolant pumps. The axial location of the pumps depends on the trade-off between the deteriorated pump performance at high coolant temperature, and the desire to eliminate vessel penetrations close to the core. The top location shown in Fig. 1 is the currently preferred position, but studies are still in progress to finalize this choice, as well as the choice of pump type. The integral vessel configuration is essential to the safety by design approach as shown in [7] and thus it is key to satisfy the enhanced Generation IV safety requirement.

To address the proliferation resistance requirement IRIS features a long life straight burn core, without shuffling or refueling, thus rendering the fuel inaccessible during operation. Key design choices are type of fuel and length of core life. Lower enriched UO_2 and MOX were the two chosen fuels because of their good proliferation resistance characteristics and the accumulated wealth of experience. Designing a straight burn long life core was a challenge; in principle, for MOX fuel (see Fig. 2) it could be attained either with a hard spectrum, tight lattice (rod pitch-to-diameter ratio of 1.1 or lower) or with a moderated, open lattice arrangement (rod pitch-to-diameter ratio of 1.8 or higher), whereas for UO_2 an open lattice is preferable (Fig. 3). An open lattice was the chosen configuration because it yields acceptable reactivity coefficients (especially void) throughout life and requires a lower fissile content UO_2 is the US preferred fuel because of non-proliferation policy, while MOX has somewhat better performance characteristics and it is of interest to the IRIS international partners. To promote proliferation resistance for the MOX fuel alternative, the use of reactor-grade Pu from highly burnt PWR fuel (> 45,000 MWd/tU) is envisioned. This fuel has a < 60% initial Pu-239 isotopic fraction, which is further reduced to less than half after burnup in IRIS. Combined with core inaccessibility during operation and effective national safeguards at delivery and retrieval, proliferation resistance is significantly enhanced in IRIS.

The lifetime reactivity control is managed primarily through integral fuel burnable absorbers (IFBA) consisting of a thin coating of ZrB_2 on the pellet. Preliminary calculations indicate that for MOX fuel the reactivity swing can be reduced from 25% to a manageable 7%. For UO_2 fuel, the initial excess reactivity is larger (~ 50%), but a proper choice of burnable poisons is expected to reduce it to about 10% to 15%.

A fissile U^{235} enrichment level of 8 to 10% yields a core lifetime of 8 to 10 years, with a discharge burnup (average) in excess of 70,000 MWd/t. This is beyond the current database, hence the fuel performance must be demonstrated as part of the licensing process. Also requiring special licensing is the level of U^{235} enrichment which is higher than in current reactors and fabrication facility licensing base. Thus, fabrication facilities must be enhanced to handle the higher enriched fuel and must be relicensed. All of the above is achievable within current technology and both the technical and licensing aspects can be positively addressed. However, the required implementation time would impact the IRIS objective of deployment within a decade. A core utilizing 4.95% enriched UO2 with a discharge burnup of ~ 40,000 MWd/t yields a straight burn core life of five years and is completely within current technical state-of-the-art and licensing status. Therefore this is the reference core for the first-of-a-kind IRIS.

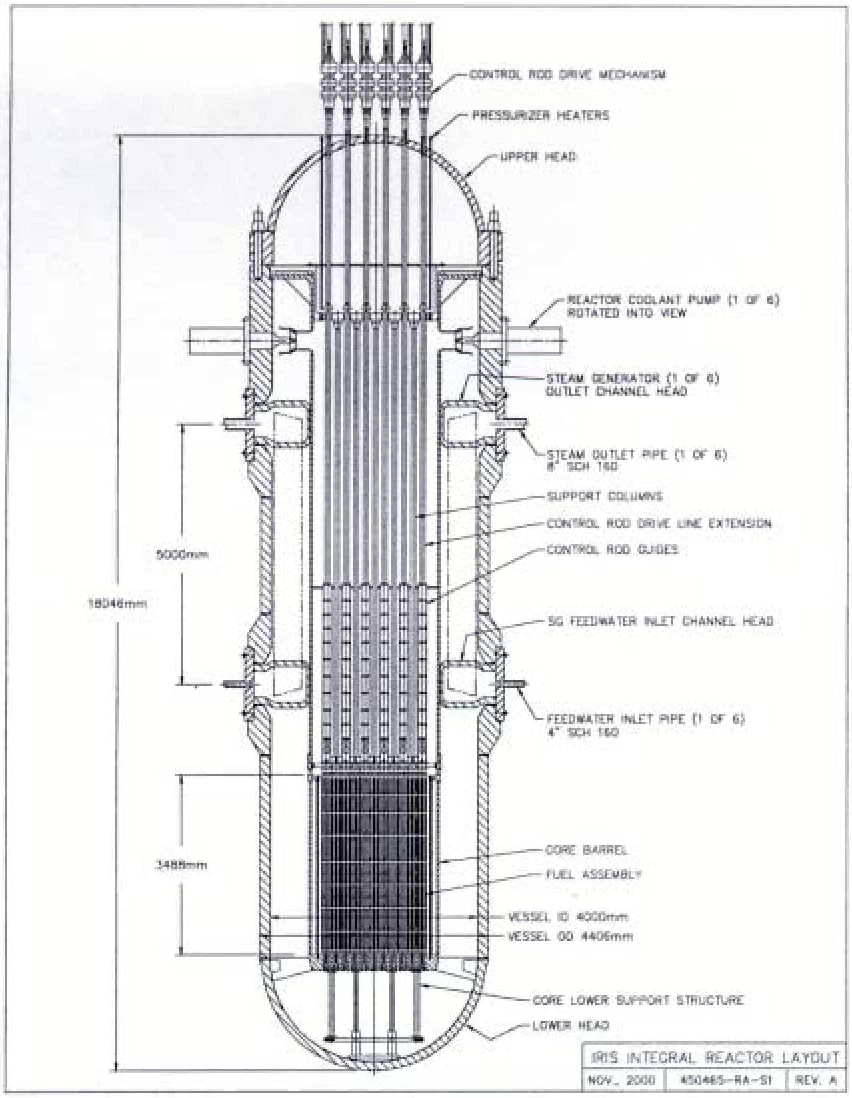

FIG. 1 IRIS Vessel Layout

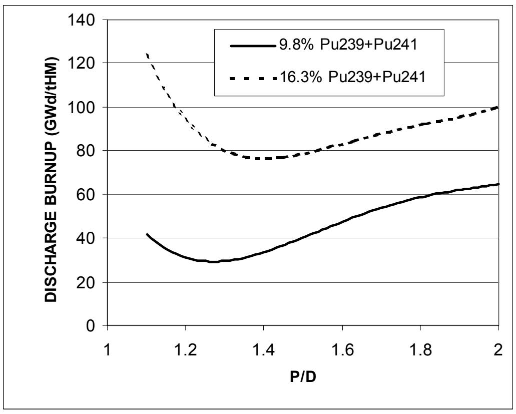

FIG. 2 Both Very Tight or Very Open Lattices Yield Higher Discharge Burnup for MOX Fuel

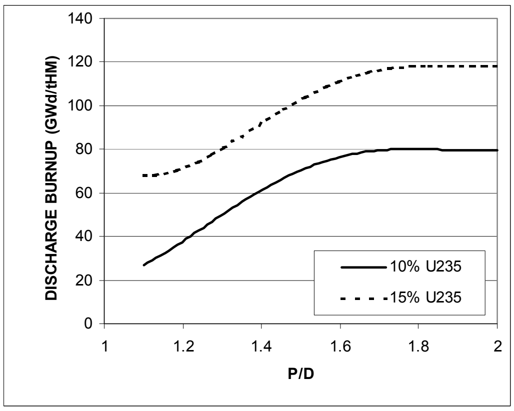

FIG. 3 Open Lattices Yield Higher Discharge Burnup for UO_2 Fuel

TABLE I. IRIS TEAM PARTNERSHIP

Team Member	Leader	Status	Scope
Westinghouse Electric, USA (W)	M. D. Carelli	Original proposal team member	Overall and tasks coordination, leadership and interfacing Nuclear island.
Polytechnic Institute of Milan, Italy (POLIMI)	C. V. Lombardi	Original proposal team member	In-vessel thermal hydraulics, steam generators, decay heat removal system, containment
Massachusetts Institute of Technology, USA (MIT)	N. E. Todreas	Original proposal team member	Core thermal hydraulics, novel fuel rod geometries, safety, maintenance
University of California at Berkeley, USA (UCB)	E. Greenspan	Original proposal team member	Core neutronics design
Japan Atomic Power Company, Japan (JAPC)	K. Yamamoto	Joined September 99	Maintenance
Mitsubishi Heavy Industries, Japan (MHI)	A. Nagano	Joined October 99	Steam generators, maintenance, modularization
Commissariat a l'Energie Atomique, France (CEA)	P. Dumaz	Joined October 99 Withdrew October 00	Core neutronics, in-vessel thermal hydraulic, safety, licensing
British Nuclear Fuels, UK (BNFL)	K. Miller	Joined November 99	Fuel and fuel cycle, cost evaluation
Tokyo Institute of Technology, Japan (TIT)	H. Ninokata	Joined January 00	Novel fuel rod geometries, detailed 3D T&H subchannel characterization, PSA
Bechtel Power Corp., USA (Bechtel)	J. R. Robertson	Joined May 00	Balance of plant, cost evaluation, construction schedule
University of Pisa, Italy (UNIPI)	F. Oriolo	Joined August 00	Containment, transient analyses
Ansaldo, Italy	G. Proto	Joined December 00	Steam generators, reactor systems
National Institute Nuclear Studies (ININ), Mexico	G. Alonso	Joined January 01	Core neutronics
NUCLEP, Brazil	M. M. Moraes	Joined February 01	Vessel, heat exchangers

IRIS is being designed to be able to accommodate interchangeably all the three cores: less than 5% enriched UO_2, about 10% enriched UO_2, and MOX. This is accomplished by keeping the outer envelope dimensions and changing the fuel rod diameter and rod pitch-to-diameter ratio (p/d); for example a MOX core with p/d ~ 1.7 can replace a UO_2 core with p/d ~ 1.4.

A straight burn core with infrequent refuelings has the potential of increasing the plant capacity factor. However, the repair and maintenance shutdowns will then become limiting. Therefore, IRIS is designed for maximum reliability, including redundancy, of components and for optimized maintenance, including on-line monitoring and maintenance. The IRIS objective is to limit the maintenance shutdown to about five-year intervals, thus matching the refueling interval for the lower enrichment core and corresponding to about half of the core life in the other cases.

The present schedule calls for completion of the preliminary design by the end of 2002, completion of the Safety Analysis Report by 2005, attainment of US NRC Design Certification by 2007 and being ready for deployment by 2010.

The small IRIS module does not have the economy of scale of LWRs large single units. However, cost reduction is realized through plant simplification (reduced piping, elimination of separate pressurizer and steam generator shells, no in-vessel refueling system, and reductions of safety-grade systems thanks to the enhanced safety features of the IRIS design) and the serial manufacturing of a large number of standardized simpler and smaller identical units (customized mass production). A preliminary economic analysis has indicated that IRIS generation costs are competitive with other nuclear and non-nuclear systems over a broad range of IRR (Internal Rate of Return) values, with a projected total cost of electricity at about 3¢/KWh.

3. THE INTEGRATED INTERNATIONAL APPROACH

When Westinghouse started the conceptual design of a new reactor to answer the DOE solicitation for what have later been called Generation IV nuclear energy systems, our overriding objective was to develop a commercially viable concept and to avoid its becoming one more paper reactor like so many of its predecessors. It was evident that the era of a single company, or even a single nation, developing and deploying a nuclear plant was past gone, not to return. Also, it was apparent that utilities are not willing to build again large plants with a price tag of a billion dollars or more. Larger plants however had economy of scale and a new dimension had to appear for smaller plants to become more economical and better market competitors.

In addition to being simpler to construct and operate, these smaller plants have to be fabricated in series, i.e. as the former American Nuclear Society president Stan Hatcher once said “we have to build aircraft, not aircraft carriers”. To fabricate and deploy an economically large enough number of multiple, identical modules, the market has to be one global, international arena; e.g. selling and installing within a reasonably short period of time a few modules in developing country A, several modules for a large station in developed country B, a single module in country C which never had nuclear power, etc.

Once it was established that this new reactor was to be deployed world-wide, it followed that to be readily accepted internationally, it had to be developed internationally, i.e. it had to address international requirements, needs and even cultures. Thence the IRIS approach, as emphasized by the first letter (International) of its acronym: IRIS was going to be designed from the very beginning and subsequently fabricated and deployed by an international partnership, where all team members were stakeholders in the project.

The approach immediately found a positive resonance, as the IRIS team kept growing from the initial four members and two countries to the present 13 members from six countries. The original team of Westinghouse, two American universities (University of California Berkeley (UCB) and Massachusetts Institute of Technology (MIT)) and one Italian university (Polytechnic of Milan (POLIMI)) was joined by Mitsubishi Heavy Industries (MHI), Japan Atomic Power Company (JAPC) and Tokyo Institute of Technology (TIT) from Japan, British Nuclear Fuel Ltd. (BNFL) from UK, Bechtel from USA, Ansaldo and University of Pisa (UNIPI) from Italy, National Institute for Nuclear Studies (ININ) from Mexico and Nuclear Heavy Components (NUCLEP) from Brazil. The team thus includes reactor vendors and component manufacturers (Westinghouse, MHI, Ansaldo, NUCLEP), fuel and fuel cycle vendor (BNFL), architect engineer (Bechtel), utility (JAPC), universities (UCB, MIT, POLIMI, TIT, UNIPI), and laboratory (ININ). The French Atomic Energy Commission (CEA) actively participated in the IRIS program until October 2000, when it decided not to be a formal member of IRIS, even though it was open to cooperation in specific areas, such as safety [7]. Other organizations have expressed interest and may join soon. Table I provides a summary of the IRIS team partnership with the areas of responsibility of each team member.

4. IRIS CHARACTERISTICS FOR AN INTERNATIONAL DEPLOYMENT

In this section it will be briefly examined how the IRIS technical and programmatic characteristics are consistent with an international deployment.

First of all, IRIS has a very flexible configuration with a design which can accommodate a wide range of module powers from less than 100 to over 300 MWe.* The capability of a wide power range in a modular configuration allows the deployment of IRIS according to the various countries needs, as mentioned in the previous section.

IRIS is based on proven LWR technology, re-engineered for improved performance. No new technology development is required; moreover, LWRs are the only reactor type with worldwide diffusion and acceptance.

The progressive sequence in core designs was chosen also taking into consideration the international deployment of IRIS. The first core based on current technology, will be part of the licensing application to the US NRC and will fuel the first-of-a-kind reactor. The proliferation resistant, long life, higher (8-10%) enrichment UO_2 and MOX cores which will be ready for deployment around 2015 are expected to be the core choices for international deployment. Because of the straight burn fuel design and to address non-proliferation considerations, it is envisioned that the IRIS consortium will deliver the new core loading and take back the spent one (preliminary evaluations indicate that for the 8-year core, only one core at a time stays in the decay pool at the site). The simple, compact design and the very infrequent maintenance required by IRIS drastically reduce the plant staff (both number and expertise) requirements. The IRIS consortium not only plans to take back the spent fuel atcore changeovers, but also the entire reactor as well, at the end of its scheduled life. Doing the latter is greatly aided by the fact that the integral vessel configuration lends itself to shielding

* A model with full natural circulation for powers less than 50 MWe has also been considered and scoped out.

of the reactor vessel by the water surrounding the core and by dedicated internal shields. Consequently, the vessel outside surface is barely radioactive. The IRIS consortium "take back" policy will greatly reduce the decommissioning required on the part of the host country. All of the above: absence of large spent fuel storage, reduced staff requirements and largely simplified decommissioning, contribute to requiring only a very limited infrastructure. Thus, IRIS can be equally deployed in developed countries familiar with nuclear power as well as in countries building their very first plant.

Obviously, a critical step in the deployment of any nuclear reactor is its licensing. In the case of the international IRIS this could mean lengthy and expensive regulatory review and approval by each country of deployment. This can be avoided by a two-step approach: a first, major and one-time-only step which applies to all countries and a second, minor step, which is country dependent. It is our goal that the first step will carry the largest portion of the effort, while the second step can be limited to minor adjustments. To fulfill the first step requirements, IRIS will be first design certified by the US NRC (the process has already started in early 2001). The IRIS design will address as much as possible other countries requirements, for example, consideration is given to implement a Core Melt Exclusion Strategy, as discussed in the companion paper in this symposium [7]. We will also pursue review and endorsement by the International Atomic Energy Agency. Even though IAEA is not a legal licensing authority, its "approval" of IRIS will carry a tremendous weight and, coupled with the US NRC design certification, will allow IRIS to be very close to its objective of "universal" licensing.

Another area where IRIS will need IAEA support is in the definition of the spent fuel final repository. An international repository will be required for IRIS spent fuel and radioactive components by virtue of the international nature of IRIS. It is easy to predict that the technical obstacles will be dwarfed by the legal and political ones, and here again IAEA will provide the most effective forum for successfully resolving these issues.

CONCLUSIONS

IRIS is an innovative reactor design which offers great promise for the new era of nuclear power. It is the first ever attempt to internationally develop a nuclear reactor system from cradle to grave. Even though the difficulties and logistics are far from being minor, the open minded and enthusiastic support of the team members, bodes well for the ultimate success of IRIS.

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DESIGN OF PASSIVELY SAFE SMALL REACTOR FOR DISTRIBUTED ENERGY SYSTEM AND TECHNOLOGY DEVELOPMENT

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Abstract

Japan Atomic Energy Research Institute (JAERI) has designed small-integrated reactors for supplying district heating or electricity to small grids. An innovative advanced marine reactor MRX aimed primarily for ship propulsion is applied to small grid electricity supply or to electricity and heat co-supply, by installing it on a barge. By extending the design concept of the MRX, a reactor for exclusive use of heat supply to households or offices is studied, which is to be sited deep underground. For these reactors, an in-vessel type control rod drive mechanism is adopted for safety enhancement and to realize a simple and compact reactor. JAERI has developed a highly reliable INV-CRDM, which is driven by an electric motor and capable of controlling finely the reactor power.

1. INTRODUCTION

From the view point of addressing global warming and energy security, it is necessary to increase the utilization of nuclear energy for not only electricity generation by large scale nuclear power plants, but also other usages such as heat supply to air conditioning, sea water desalination etc. A “Long-term Program for Research, Development and Utilization of Nuclear Energy in Japan” established in fiscal year 2000 by Japanese Atomic Energy Commission proposes that more innovative reactor technologies including reactors for heat supply should be developed.

Japan Atomic Energy Research Institute (JAERI) has been designing small reactors for a distributed energy supply system such as district heat supply systems, small grid electricity supply systems, and so forth. In the present stage, two systems are studied. One is a small electricity generation plant and seawater desalination system using the advanced marine reactor MRX (Marine Reactor X [1]) installed on nuclear barge, and the other is a district heat supply system with a reactor of exclusive use for heat supply sited close to a demand area. While the MRX is of the basic design stage, the latter is a preliminary design, which is on the base of the MRX design concept called as PSRD (Passively Safe small Reactor for Distributed energy system). The PSRD, which is to be sited deep underground in the present study, is designed to supply heat for air conditioning or hot water for household or office uses.

One of the key technologies to be developed for the realization of these reactor designs is an in-vessel type control rod drive mechanism (INV-CRDM) installed inside the reactor vessel, which can eliminate the possibility of a rod ejection accident and allows the reactor to be compact and simple. JAERI has completed the development of a highly reliable INV-CRDM, which is driven by the electric motor and capable of controlling finely the reactor power. This paper describes the design of power plants installed on barges, the district heat supply system with the exclusive reactor, and the development of the INV-CRDM.

2. POWER PLANTS INSTALLED ON BARGES WITH MRX

Two types of the nuclear barge are studied – one is for electricity generation, the other is for an electricity and fresh water co-generation system. Both of the systems adopt the advanced marine reactor MRX. The technical characteristics of MRX is described in detail in the paper [1] and it will not be repeated here, but a brief description will be given in section 2.1 for convenience of the reader, focusing on safety and system simplicity.

2.1. MRX

In order to achieve the design goals of being light-weight, compact, simple and safe, MRX adopts the following design improvements.

i) an integral type reactor,

ii) the INV-CRDM,

iii) a water-filled containment,

iv) a passive decay heat removal system, and

v) one-piece removal of the reactor system..

1. Integral type reactor

A cross section of the reactor pressure vessel together with that of the containment vessel is shown in Fig. 1. Effective layout of the primary components makes the reactor compact by installing the most of components inside the vessel: The core is located in the lower part, the steam generator in the middle part, the CRDMs and pressurizer in the upper part inside of the reactor pressure vessel, and the primary coolant pumps are connected directly to the flange of this vessel. The main specifications of the core, the CRDMs, the SGs, the pressurizer, and the primary coolant pumps are shown in Table 1.

The steam generators (SGs) of the MRX are of the once-through, helical coil tube type being suitable for the integral type reactor. The primary cooling water flows outside of the tubes, and the secondary water and steam flow inside the tubes. The pressurizer of ringed flatness type is placed outside the INV-CRDM to use the space effectively in the RPV. The pressurizer comprises the heater, the spray, the relief valves, and the safety valves. Water inside the pressurizer passes to the hot leg through surge holes at the bottom of it.

2. Reactor core

Since the MRX uses control rods for power control and eliminates chemical shim, the parameters of the reactor physics are characterized by a rather large negative moderator density reactivity coefficient (α_m) for the core cycle life. The large value of α_m can contribute effectively to self-regulating reactor control property even for a heavy load change. On the other hand, each control rod cluster has a large value of control reactivity, which is a disadvantage for a possible rod ejection accident (REA). For the REA, however, adoption of the INV-CRDM can eliminate the possibility of this accident.

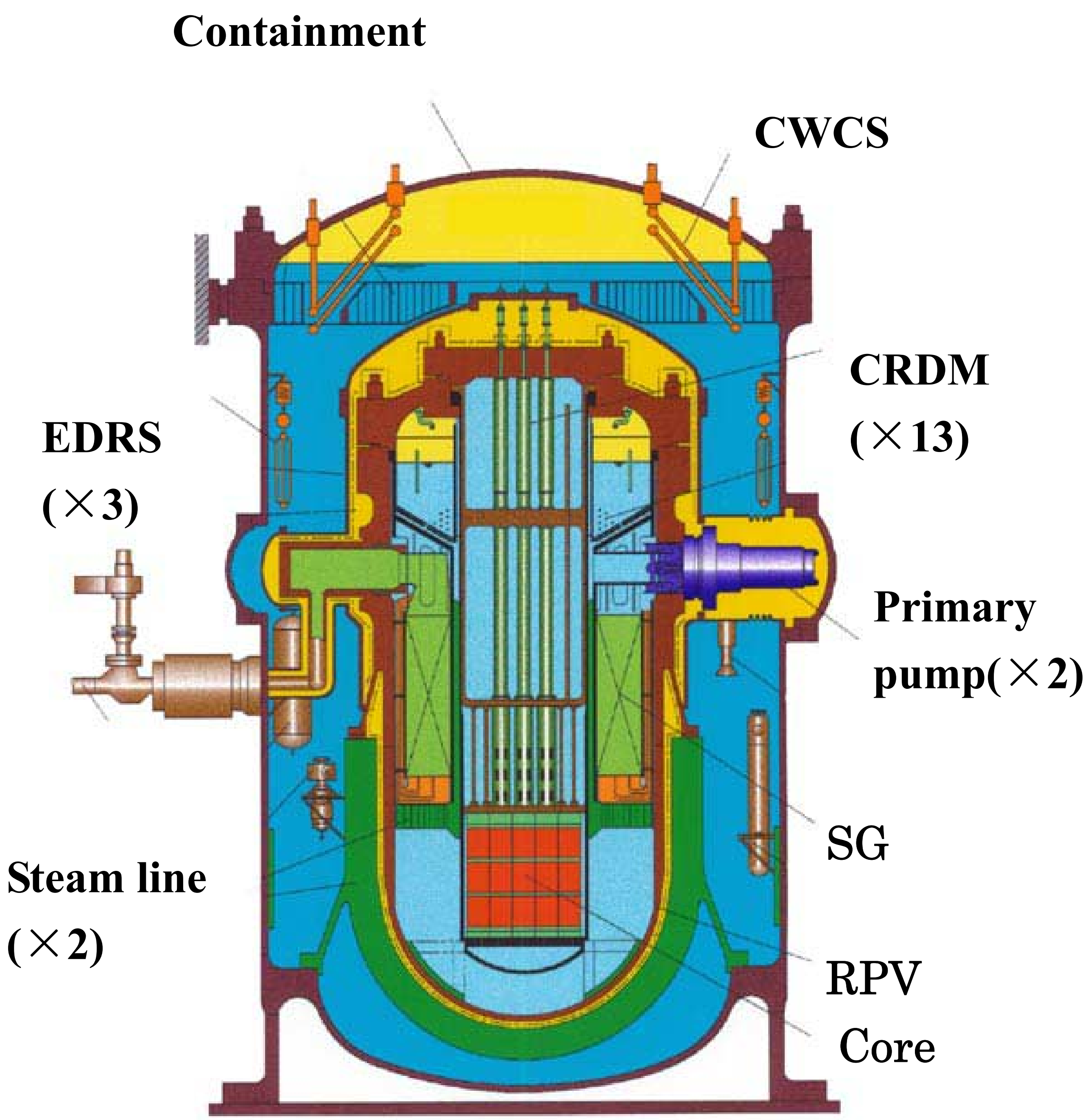

FIG. 1. MRX.

The core power profile of the MRX is not as uniform as that adopting the chemical shim. To flatten the power profile, MRX adopts fuel rods with Gd_2O_3 and the burnable poison rods (BPs) filled up with boron glass, and it increases the number of control rods. Although, the total peaking factor of the core power distribution is relatively large (3.98), the maximum of linear heat rate is 30.4 kW/m which has enough margin to the limiting value, 41kW/m for fuel rod designing. To obtain a high burn-up without increasing the excess reactivity at the beginning of core cycle, a two-batch fuel shuffling strategy is adopted, in addition to the adoption of the fuel rods and BPs mentioned above. Refueling will be able to be performed on a dockyard or land based facility on the same period of the mandatory hull survey per four years. The average burn-up is 23GWD/t and the life of core cycle is eight years, by assuming the core load factor of 50%.

TABLE1 MAJOR PARAMETERS OF MRX

Reactor Power	100 MWt	Main coolant pump	
Reactor type	Integral type	Type	Horizontal axial flow canned motor type
Reactor coolant			
Operating Pressure	12 MPa	Rated power	200kW
Inlet/Outlet Temp.	282.5/297.5 ℃	No. of pumps	2
Flow rate	4,500 t/h	Steam generator	
Core/Fuel		Type	Once-through helical coil type
Equivalent Dia	1.49 m	Tube material	Incoloy 800
Effective height	1.40 m	Tube outer/inner dia.	19/14.8 mm
Ave. linear heat flux	7.9 kW/m	Steam temp./press.	289℃/4.0 MPa
Fuel type	Zry-clad UO2 fuel	Steam flow rate	168 ton/h
U-235 enrichment	4.3%	Heat transfer area	754 m^2
Fuel inventory	6.3 ton	Reactor vessel	
Fuel Ave. burn-up	22.6 GWd/t	Inner dia./height	3.7/9.7 m
No. of fuel assembly	19	Containment	
Fuel rod outer dia.	9.5 mm	Type	Water-filled RV immersion type
Control rod drive mechanism		Inner dia./height	7.3/13 m
Type	In-vessel type	Design press.	4 MPa
No. of CRDM	13		

3. In-vessel type control rod drive mechanism, INV-CRDM

The INV-CRDM works in the primary loop water of which condition is very severe, the high temperature and pressure (583K,12 MPa). Details of the INV-CRDM will be presented in later section.

4. Water-filled containment

The functions of the water-filled containment are to passively maintain core flooding in case of accidents including a LOCA, to shield against radiation, as well as to enclose the area for prevention of radioactive materials release to the surroundings. There is nitrogen gas in the upper space over the water surface of the containment. Core flooding can be maintained passively by pressure balance of the containment and the pressure vessel, and with the help of an emergency decay heat removal system (EDRS) and an emergency containment water cooling system (ECWCS). Thus, core flooding in a LOCA can be attained passively without the emergency core cooling pumps (ECC pumps) or an accumulator, if the containment initial water level is appropriate. Experiments[2] on the thermal hydraulic behavior of water-filled containments was conducted and the relationships between the initial water level of containment and the balance pressure, etc., were obtained, and the principal function of core flooding was confirmed.

Inside the containment, the RPV, the primary loop piping, etc., are installed in the water. Therefore, thermal insulation is necessary to prevent the heat loss from the surface of these components into the water. The RPV is covered with a water-tight shell made of stainless steel of 45 mm thickness. Between the RPV and the water-tight shell, stainless steel felt insulation is inserted. The heat loss from the RPV with this insulation is estimated to be less than 1% of the rated power. Water inside the containment has also a role of radiation shielding, which can eliminate the concrete shield outside the containment. This merit makes the MRX plant drastically light in weight. For shielding, in addition to the water inside the containment, a steel shield is inserted between the core and the steam generator, and a cast steel shield of 45 cm thickness is set outside the RPV, together with making the core barrel thick.

5. Passive decay heat removal system

When the normal procedure for decay heat removal through the steam generator and the residual heat removal system is not available due to an accident, decay heat is removed passively by means of the EDRS and the ECWCS; The decay heat is transferred from the primary coolant to the water of the containment through the heat exchanger of the EDRS, and from the water of the containment to the atmosphere through that of ECWCS. The flow is driven by only natural circulation forces. The EDRS begins to transfer the heat by opening the valves, but the ECWCS (of heat-pipe type) always transfers the heat because of no valve setting. For this procedure, only opening of the valve is needed, and a small power source is enough to open the valves.

6. Safety and simplicity evaluation

Safety transient analyses such as a LOCA, SGTR, etc., were conducted by using RELAP5/mod2 and COBRA-IV codes. Even in case of a tube rupture accident of the maximum diameter of 50 mm, the core can be flooded always throughout the transient. Decay heat can be removed by the EDRS.

The reliability of the reactor system is evaluated on the basis of the probabilistic safety analyses (PSA) by the event tree method on a LOCA, a SGTR, and others. The total occurrence frequency of core damage is approximately 4×10^{-8}/reactor/year. The total occurrence frequency of the core damage of the MRX is two orders of magnitude lower than that of existing PWR plants such as Surry, Sequoyah, and Zion.

System simplification is also one of the most important design goals. The systems in MRX for normal operation are almost the same as in existing PWRs except for the chemical processing system. Elimination of the chemical processing system makes the normal operation system of the MRX simpler. The engineered safety system of the MRX is simplified significantly. The number of constituent equipment for the engineered safety system are very few compared with that of existing PWRs and AP-600. The simplification of the systems leads to improvements in economy and reliability through reductions of plant construction and maintenance costs, reductions in human error during maintenance, and reduced probability of equipment failure.

2.2. Nuclear barge for electricity generation of 100 MW

A nuclear barge with the MRX can be sited in protected water area such as an artificial sea bay or river. The reactor thermal power is 334 MW to generate the electricity of 100 MW with two 167 MWt MRXs. This electricity capacity will be sufficient for household use of 100,000 persons on the base of 1 kW per person. Upgrading the power of the MRX is possible without change of the design concept. The concept of the nuclear barge is shown in Fig. 2.

Displacement of the barge is 13,000 tons. Mooring of the barge is one important design aspect and depends on the condition of the site. A possible mooring facility is made by a hydraulic shock-absorbing mechanism using a dolphin type one which can be used for large floating structures. Maintenance and refueling of the reactor system can be done by means of one-piece removal of the reactor system at a dockyard, and replacment with another one. The on-barge type plant will not require immediate evacuation of public in case of a reactor accident, because the MRX can be easily migrated from the energy consumption area.

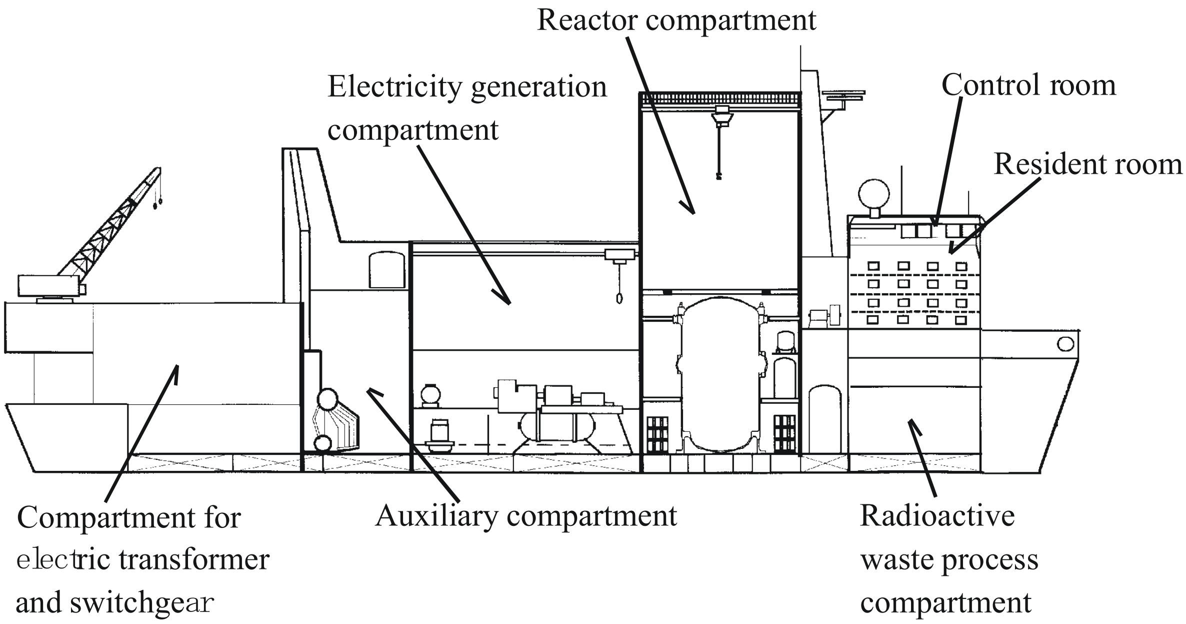

FIG. 2. Nuclear barge with MRX.

2.3. Electricity and fresh water co-generation system

With help of energy supply from the nuclear barge mentioned above, a sea water desalination system can be furnished inside the nuclear barge or a separate desalination barge as shown in Fig. 3. The RO system is selected by taking account the capability of changing easily the supply ratio of electricity and fresh water according to demand. That is, the electricity generated can be used variably for the electricity supply or the desalination.

The desalination capability of the RO system is 4 to 7 $kWe{\cdot}h/m^3$ depending on the seawater salinity and temperature, recovery ratio, required permeate quality, plant configuration and implementation of energy recovery in the brine blow down. It is said that the RO system can produce fresh water with total dissolved solids (TDS) of 100 to 200 ppm depending on the system configuration and the feedwater salinity. For requirement of lower TDS, a two-stage configuration will be needed to get higher separation performance. In the present study, brine conversion two stage RO seawater desalination system is applied.

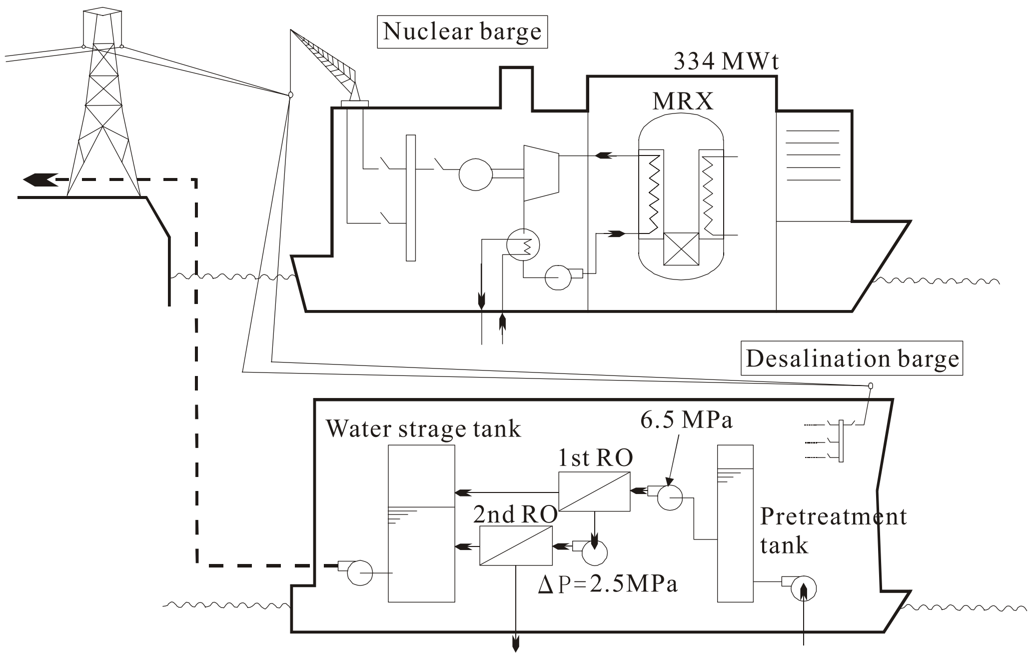

FIG. 3. Flow diagram of electricity generation and desalination.

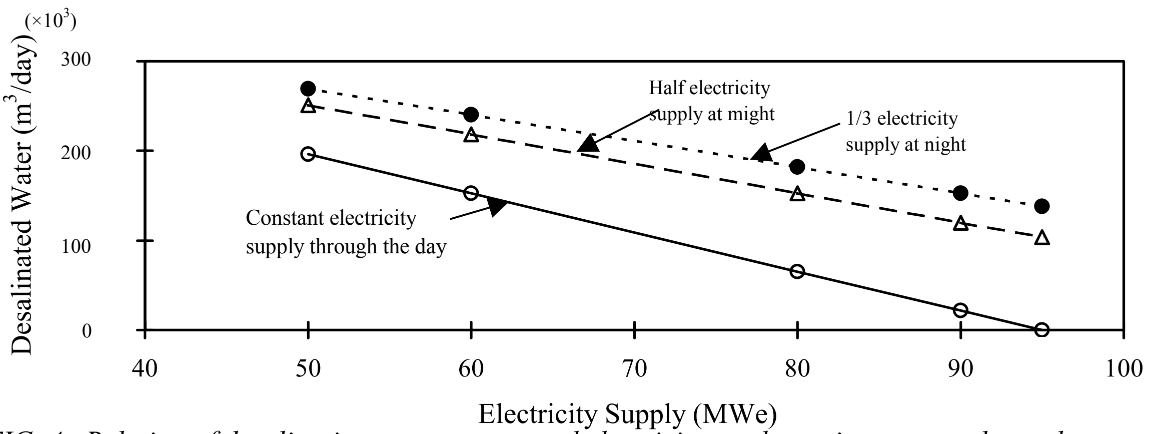

FIG. 4. Relation of desalination water output and electricitysupply at given ractor thermal output of 333MW.Electricity supply at the night time (12 hrs) is taken to decrease and the surplus is used for desalination.

The relation of desalination water output and electricity supply with the 334 MW of reactor output is presented in Fig. 4. For an electricity supply of 50 MW through the daytime, the fresh water of 200,000m^3/day can be supplied by taking account on electricity consumption of 5 MW in the barge. This amount of fresh water will be sufficient for household use of 400,000 persons on the average consumption base of 0.5 $m^{3/}$day/person. At night time, demand of the electricity at household use will decrease in general. This surplus electricity can be consumed effectively to the desalination. Two cases are considered for demand reduction of electricity at the night time for reference: the supply of electricity at the night time (12 hours) decreases to half of the daytime figure (12 hours), and it decreases to a third. The results are depicted in the figure for the constant reactor power.

Additionally a further possible option is to reduce the total reactor power output at the night time following the demand decrease of both electricity and fresh water. The MRX can comply easily with this changeable demand because it has a fine self-control characteristic of the reactor power against the load change.

3. DISTRICT HEAT SUPPLY SYSTEM WITH THE EXCLUSIVE REACTOR SITED UNDER GROUND

A conceptual design of PSRD exclusive for heat supply has commenced in JAERI. Since this reactor supplies only relatively low-pressure steam, the operating pressure and temperature of the primary loop are lower than those of the MRX. In design of the PSRD, high priority is laid on enhancement of safety as well as improvement of economy, since the reactor should be to be sited close to the energy demand area. Safety enhancement can be attained by extreme reduction of probability in accident occurrence, adoption of passive safety system and natural circulation cooling of the reactor. Economic improvements can be made by simplification of the system, increase of factory-manufacturing, long-term operation of the reactor, etc. The preliminary design concept of the PSRD for heat supply with thermal output of 100 MW is shown in Fig. 5. The PSRD is an integral type reactor by installing all components inside the reactor vessel in order to decrease the possibility of LOCA and to eliminate the possibility of a rod ejection accident. The pipes penetrating the pressure vessel wall are designed to be limited to only the feed water pipes, the steam pipes and the pipe for the safety valve. These pipes are connected at the upper cover of the reactor vessel. To realize this design concept, possibility of elimination of the volume control system and adoption of emergency decay heat removal system without the penetrating pipe are now examined in a detailed study.

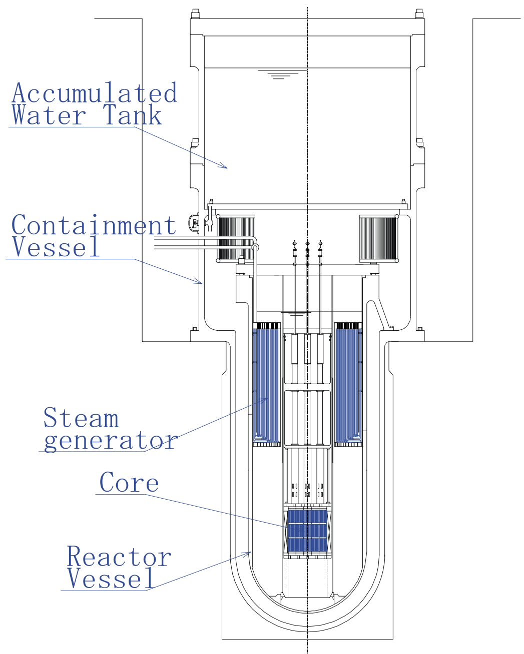

FIG. 5. PSRD

Major parameters

Thermal output(MW)	100
Core outlet temp.(℃)	233
Core pressure(MPa)	3
Core life(year)	8
No. of fuel assembly	37
Fuel arrangement	square lattice
Fuel	UO2
(	4% enrichment)
Fuel cladding	Zry-4
Steam generator	
・steam flow rate(t/h)	150
・temperature of steam/feed water(℃)	180/190
・pressure of steam(MPa)	0.88
Diameter of containment(m)	8.5
Height of containment(m)	

The PSRD core consists of 37 assemblies with Zircaloy clad UO_2 fuel rods, whose specification is the same as the one used in the 17 × 17-type fuel assembly for current PWRs. The core life cycle is estimated as 10 years with a ^{235}U concentration of about 4%, by assuming the core load factor of 50%.

The reactor is cooled by natural circulation and pressurized by self-pressurization. The temperature of core outlet is 233 °C, and the pressure is 3 MPa. The steam generators produce steam of 150 t/h, 180°C, and 0.88MPa with the feed water of 90 °C.

The space inside the containment vessel is maintained at a vacuum in normal operation for thermal shielding, preventing the heat from transferring through the space to the containment vessel. In the case of a loss of coolant accident, the water accumulated in the water tank will be emptied into the containment. The decay heat of the core can be passively removed by conducting heat through the reactor vessel wall to the water of the containment. The heat transferred to the water can be rejected through the emergency cooler in the containment to the outside of containment. The machinery working inside the reactor vessel is only the INV-CRDM, which is inevitable in the PSRD. Adoption of this makes the reactor vessel and the containment vessel compact, as well as elimination of possibility of a rod ejection accident.

The PSRD for heat supply system is installed in a cave placed at a depth of 50 m under energy consumption areas. In the case of a reactor accident, immediate public evacuation will not be required because considerable environmental impact at surface of the ground does not appear for several tens years after the accident. Reactors are automatically and remotely operated at a supervisory and control center.

4. DEVELOPMENT OF IN-VESSEL TYPE CONTROL ROD DRIVE MECHANISM

4.1. Concept of the whole structure

A conceptual drawing of the INV-CRDM is given in Fig. 6. It consists of a driven motor, a latch mechanism, ball bearings, a driving shaft, a position detector, and so forth. The main functions of the CRDM are to move the control rods vertically upward or downward, and to insert them instantly into the core for scram.

Rotational torque generated by the driving motor should be transferred into linear movement of the driving shaft. A mechanism to disconnect the driven motor from the driven shaft is necessary for a shorter scramming time. An innovative latch mechanism using separable ball nuts divided into three sections, which will be described later in section 3.3, has been developed to comply with these requirements simultaneously. In normal operation, the driving shaft can move up and down by rotation of the driving motor when the separable ball-nuts of the latch mechanism are closed to grasp the driving shaft by energizing a latch magnet. In a scram, the separable ball-nuts will open to separate the driving shaft by de-energizing the latch magnet, and the driving shaft drops rapidly by the self-weight and spring force. The reliability of these motions at the severe condition should be certified and verified through functional and durability tests.

As shown in Fig.6, the driving motor and the latch mechanism have a center hole, where the driving shaft together with a magnet of the rod position detector moves vertically. This arrangement is also effective for compactness of the system. Cables for a rod position detector signal and electric lines of the motors penetrate the reactor pressure vessel.

The major parameters are given in Table 2 and are compared with that of a typical PWR, which are of the out-vessel type CRDM. The outer diameter of the INV-CRDM of MRX is smaller than the other. The maximum required drawing force to the INV-CRDM of MRX is 2.2 kN, including vertical load of the CRDM, the total weight of control rod cluster, and a spring force (1.1 kN) helping scram at the condition of any ship posture within the required time. Regarding materials, the whole of the CRDM is made of inorganic materials, not organic material, taking account of the operating conditions under high temperature water and the radioactive rays.

4.2. Tests of performance and durability

Each component mentioned above was separately tested with a small scale or full sized ones on their performance at conditions of the room temperature or the high temperature. Following the tests of each component, the full sized components were assembled for an overall performance test and endurance test at the same condition as the MRX operation. The purpose of the tests was to confirm that it covers the required functions in the design condition.

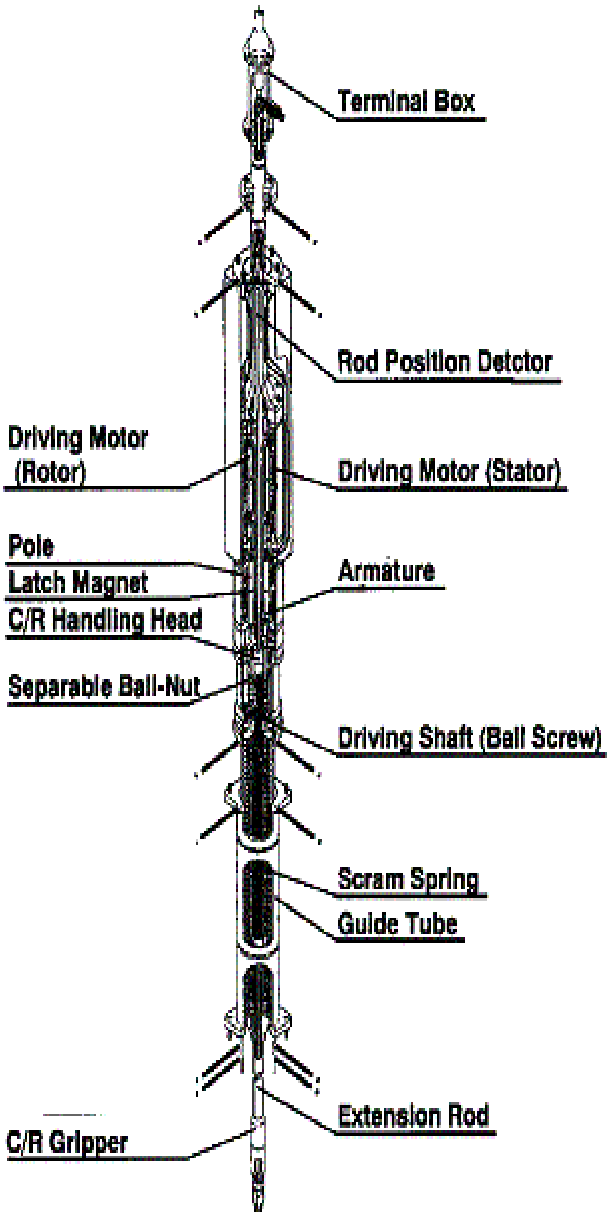

FIG. 6. CRDM

TABLE 2 MAJOR PARAMETERS OF INV-CRDM AND COMPARISON WITH CONVENTIONAL ONE

Item	MRX	PWR
Type	In-vessel, motor driven	Out-vessel, Mag-jack
Operating condition:	In the primary water (310℃, 12MPa)	Magnet coil in air (<180℃, 0.1MPa)
Dimensions :		
Outer diameter	200mm	274mm
Total lengtht	1735mm	4468mm
Stroke	1400mm	3620mm
Drive force required	2.2kN for drawing up	about 1.6kN for drawing up
Scramming force by	Weight and spring	Weight
Design operational life	20 years	40 years

With performance and durability tests mentioned above under the high- temperature and high-pressure water verified that the INV-CRDM of MRX is capable of working for the designed life period of 20 years[2].

CONCLUSION

An innovative advanced marine reactor MRX aimed primarily to use for ship propulsion can be used to supply small electricity grids or electricity and heat co-supply systems, by installing it on a barge. A nuclear barge of displacement of 13,000 tons with MRX installed with a thermal output of 334 MW can supply electricity only of 100MW or electricity of 50MW through the day-time and fresh water of 200,000 $m^{3/}$day. A reactor for the exclusive use of heat supply to household or office is studied, which is to be sited deep underground is designed in JAERI.

The JAERI has designed the in-vessel type control rod drive mechanism, which is adopted in these reactors. This enables safety enhancement and to realize a simple and compact reactor.

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CAREM: AN ADVANCED INTEGRATED PWR

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Abstract

CAREM is an Argentine project to achieve the development, design and construction of an innovative, simple and small Nuclear Power Plant (NPP). This nuclear plant has an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the design, and also contributes to a higher safety level. Some of the high level design characteristics of the plant are: integrated primary cooling system, primary cooling by natural circulation, self-pressurised primary system and safety systems relying on passive features.

1. INTRODUCTION

The Argentinean CAREM project, which is jointly developed by CNEA and INVAP, consists on the development, design and construction of an advanced, simple and small Nuclear Power Plant (NPP). The CAREM concept was first presented in March 1984 in Lima, Peru, during the IAEA conference on small and medium size reactors. CAREM design criteria, or similar ones, have been adopted by other plant designers, originating a new generation of reactor designs, of which CAREM was, chronologically, one of the first. The first step of this project is the construction of the prototype of about 27 MWe (CAREM-25). This project allows Argentina to sustain activities in the nuclear power plant design area, assuring the availability of updated technology in the mid-term [1]. The design basis is supported by the cumulative experience acquired in Research Reactor design, construction and operation, and Pressurized Heavy Water Reactors (PHWR) Nuclear Power Plants operation as well as the development of advanced design solutions [2].

2. CAREM INNOVATION

CAREM-25 is an indirect cycle reactor with some distinctive features that greatly simplify the design and also contributes to a high safety level. Some of the high level design characteristics are:

- Integrated primary cooling system.
- Primary cooling by natural circulation.
- Self-pressurised.
- Safety systems relying on passive features.

2.1. Primary system

The CAREM reactor pressure vessel (RPV) contains the core, steam generators, the whole primary coolant and the absorber rods drive mechanisms (Fig 1). The RPV diameter is about 3.2 m and the overall length is about 11 m.

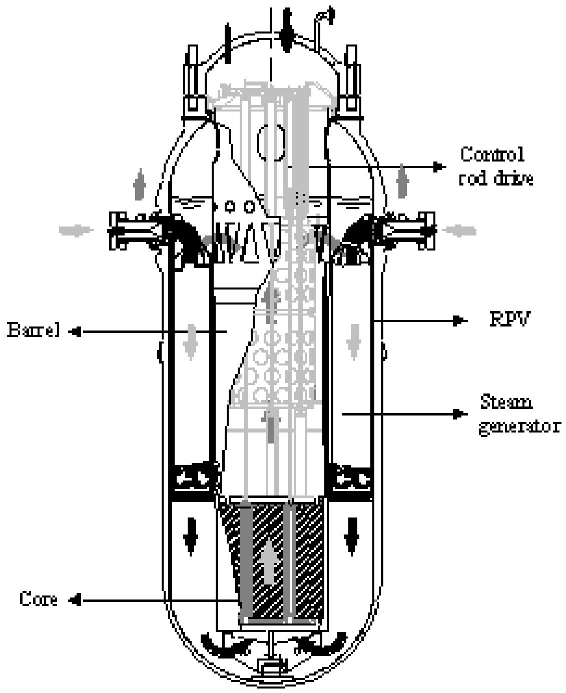


FIG. 1. Reactor pressure vessel.

The core of the prototype has 61 hexagonal cross section fuel assemblies (FA) of about 1.4 m active length. Each fuel assembly contains 108 fuel rods, 18 guide thimbles and 1 instrumentation thimble (Fig 2). Its components are typical of PWR fuel assemblies. The fuel is enriched UO_2. Core reactivity is controlled by the use of Gd_2O_3 as burnable poison in specific fuel rods and movable absorbing elements belonging to the Adjust and Control System. Chemical compounds are not used for reactivity control during normal operation. The fuel cycle can be tailored to customer requirements, with a reference design of 330 full-power days and 50% of core replacement.

Each Absorbing Element (AE) consists of a cluster of rods linked by a structural element (namely “spider”), so the whole cluster moves as a single unit. Absorber rods fit into the guide tubes. The absorbent material is the commonly used Ag-In-Cd alloy. Absorbing elements (AE) are used for reactivity control during normal operation (Adjust and Control System), and to produce a sudden interruption of the nuclear chain reaction when required (Fast Shutdown System).

Twelve identical ‘Mini-helical’ vertical steam generators, of the “once-through” type are placed equally distant from each other along the inner surface of the Reactor Pressure Vessel (RPV) (Fig 3). They are used to transfer heat from the primary to the secondary circuit, producing dry steam at 47 bar, with 30°C of superheating.

The location of the steam generators above the core produces natural circulation in the primary circuit. The secondary system circulates upwards within the tubes, while the primary goes in counter-current flow. An external shell surrounding the outer coil layer and adequate seal form the flow separation system. It guarantees that the entire stream of the primary system flows through the steam generators.

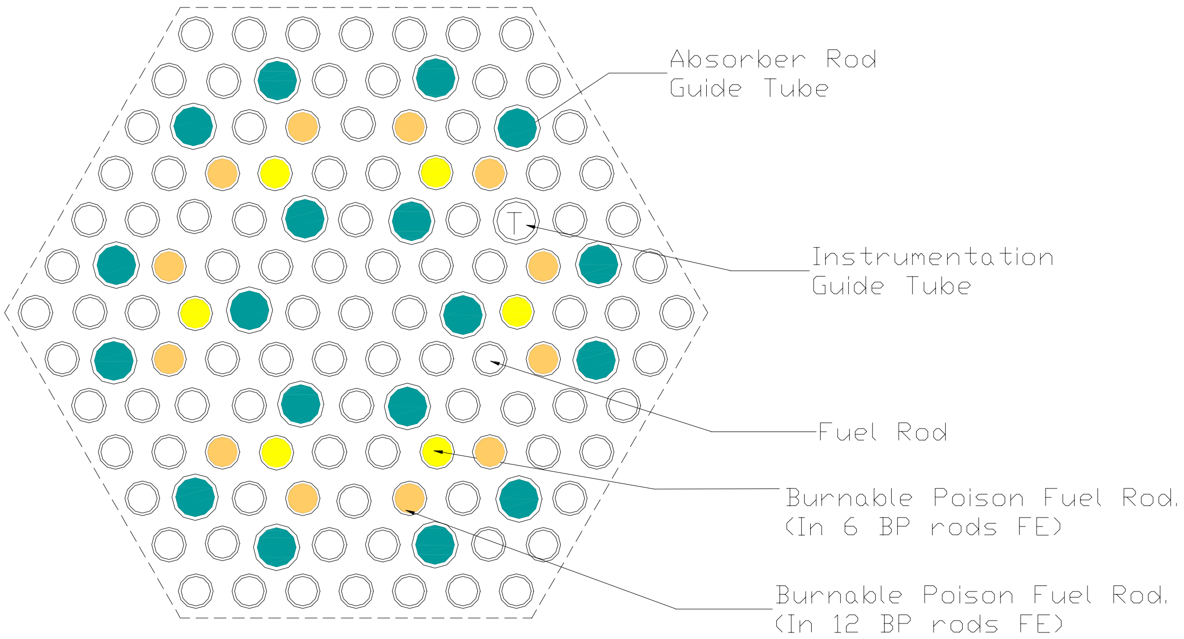


FIG. 2. Fuel Assembly diagram. Fuel rods, guide thimbles and instrumentation thimble distribution.

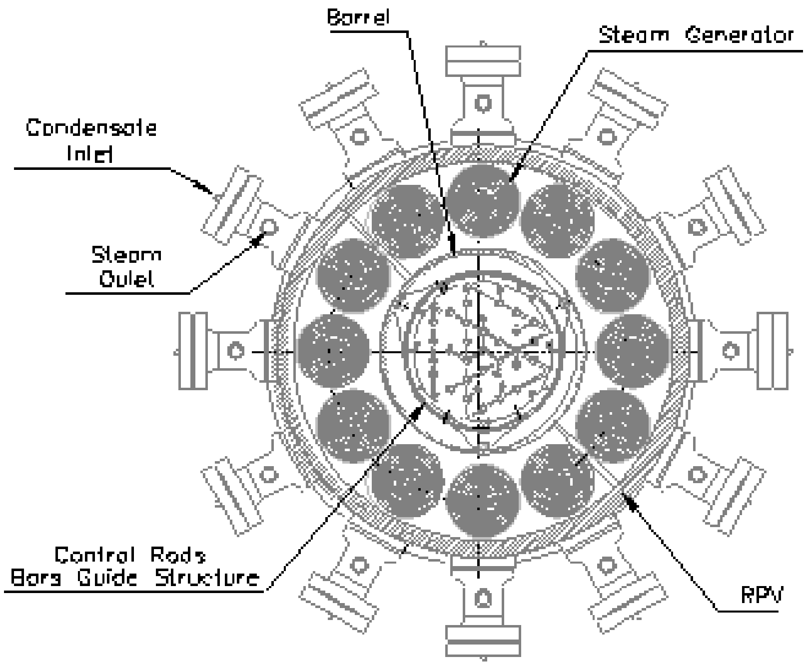


FIG. 3. Steam Generation lay out.

In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized by changing the number of tubes per coil layer. Thus, the outer coil layers will hold a larger number of tubes than the inner ones. Due to safety reasons, steam generators are designed to withstand the primary pressure without pressure in the secondary side and the whole live steam system is designed to withstand primary pressure up to isolation valves (including the steam outlet/water inlet headers) for the case of SG tube brake. The natural circulation of the coolant produces different flow rates in the primary system according to the power generated (and removed). Under different power transients a self-correcting response in the flow rate is obtained [3].

Due to the self-pressurising of the RPV (steam dome) the system keeps the pressure very close to the saturation pressure. At all the operating conditions this has proved to be sufficient to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in the case of power ramps. The negative reactivity feedback coefficients and the large water inventory of the primary circuit combined with the self-pressurisation features make this behaviour possible with minimum control rod motion. It concludes that the reactor has an excellent behaviour under operational transients.

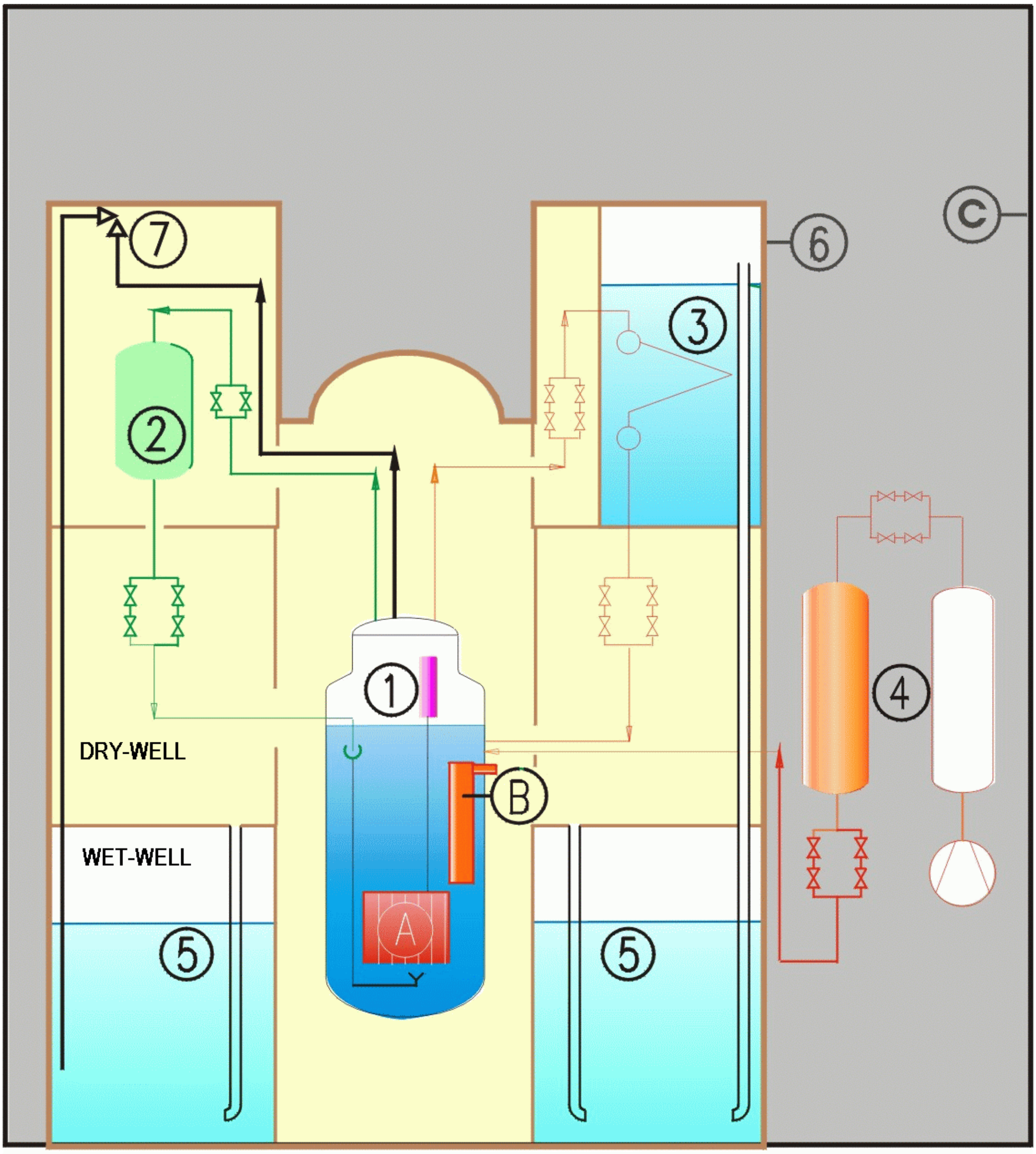
2.2. Safety systems

CAREM safety systems are based on passive features and must guarantee no requirement for active actions to mitigate accidents for a long period of time (Fig 4). They are duplicated to fulfil the redundancy criteria. The shutdown system should be diversified to fulfil regulatory requirements.

The First Shutdown System (FSS) is designed to shut down the core when an abnormality or a deviation from normal situations occurs, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping a total of 25 neutron-absorbing elements into the core by the action of gravity. Each neutron absorbing element is a cluster composed of a maximum of 18 individual rods which are together in a single unit. Each unit fits well into guide tubes of each fuel assembly.

Hydraulic Control Rods Drives (CRD) avoid the use of mechanical shafts passing through the RPV, or the extension of the primary pressure boundary, and thus eliminates any possibilities of big Loss of Coolant Accidents (LOCA) since the whole device is located inside the RPV. Their design is an important development in the CAREM concept [4]. Six out of twenty-five CRD (simplified operating diagrams are shown in Fig 5) are the Fast Shutdown System. During normal operation they are kept in the upper position, where the piston partially closes the outlet orifice and reduces the water flow to a leakage. The CRD of the Adjust and Control System is a hinged device, controlled in steps fixed in position by pulses over a base flow, designed to guarantee that each pulse will produce only one step.

Both types of device perform the SCRAM function by the same principle: “rod drops by gravity when flow is interrupted”, so malfunction of any powered part of the hydraulic circuit (i.e. valve or pump failures) will cause the immediate shutdown of the reactor. The CRD of the Fast Shutdown System is designed using a large gap between piston and cylinder in order to obtain a minimum dropping time thus taking few seconds to insert absorbing rods completely inside the core. For the Adjust and Control System CRD manufacturing and assembling allowances are stricter and clearances are narrower, but there is no stringent requirement on dropping time.



SAFETY SYSTEMS

1 - FIRST SHUT-DOWN SYSTEM

2 - SECOND SHUT-DOWN SYSTEM

3 - RESIDUAL HEAT REMOVAL SYSTEM (EMERGENCY CONDENSER)

4 - SAFETY INJECTION SYSTEM

5 - SUPPRESSION POOL

6 - CONTAINMENT

7 - PRESSURE RELIEF SYSTEM

REFERENCES

A - CORE

B - STEAM GENERATOR

C - BUILDING (SECONDARY CONTAINMENT)

FIG. 4. Containment and safety systems.

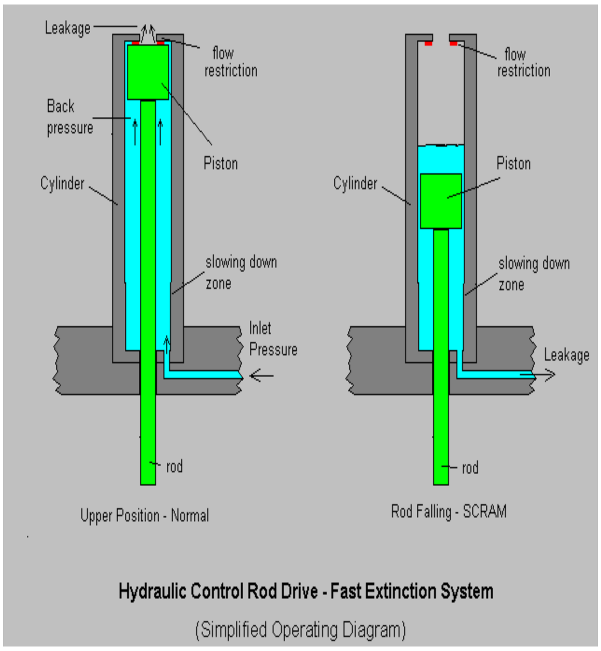


FIG. 5. Simplified operating diagram of a hydraulic control rod drive (Fast Shutdown System).

The second shutdown system is a gravity-driven injection device of borated water at high pressure. It actuates automatically when the Reactor Protection System detects the failure of the First Shutdown System or in case of LOCA. The system consists of two tanks located in the upper part of the containment. Each of them is connected to the reactor vessel by two piping lines: one from the steam dome to the upper part of the tank, and the other from a position below the reactor water level to the lower part of the tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharge of a single tank produces the complete shutdown of the reactor.

The residual heat removal system has been designed to reduce the pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensate. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and is condensed on the cold surface of the tubes. The condensate is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water is then condensed in the suppression pool of the containment.

The Emergency Injection System prevents core exposure in the event of LOCA. In the event of such an accident, the primary system is depressurised with the help of the emergency condensers to less than 15 bar, with the water level over the top of the core. At 15 bar a low pressure water injection system comes into operation. The system consists of two tanks with borated water connected to the RPV. The tanks are pressurized, thus when during a LOCA the pressure in the reactor vessel reaches 15 bar, the rupture disks break and the flooding of the RPV starts.

Three safety relief valves protect the integrity of the reactor pressure vessel against overpressure, in case of strong unbalances between the core power and the power removed from the RPV. Each valve is capable of producing 100% of the necessary relief. The blow-down pipes from the safety valves are routed to the suppression pool.

The primary system, the reactor coolant pressure boundary, safety systems and high-pressure components of the reactor auxiliary systems are enclosed in the primary containment - a cylindrical concrete structure with an embedded steel liner. The primary containment is of the pressure-suppression type with two major compartments: a drywell and wetwell. The drywell includes the volume that surrounds the reactor pressure vessel and the second shutdown system rooms. A partition floor and cylindrical wall separate the drywell from the wetwell. The lower part of wetwell volume is filled with water that works as the condensation pool, and the upper part is a gas compression chamber.

For CAREM-25 accident analysis several initiating events were considered. They were grouped into Reactivity Insertion, Loss of Heat Sink (LOHS) and Loss of Coolant Accident (LOCA) [5]. As there are no primary pumps Total Loss of Flow Accident (LOFA) is not applicable in this case.

As a general conclusion after the accident analysis, it could be said that, due to the large coolant inventory in the primary circuit, the system has large thermal inertia and long response time in case of transients or severe accidents.

2.3. Advantages of the CAREM design

Technical and economical advantages are obtained with the CAREM design compared to the traditional design:

- No large LOCA has to be handled by the safety systems due to the absence of large diameter piping associated to the primary system. The size of maximum possible break in the primary is 38 mm.
- The rod ejection accident has also been eliminated due to the development of innovative hydraulic mechanism completely located inside the reactor pressure vessel. Furthermore, the hydraulic control rod drive mechanism has a significantly lower cost compared with current PWR control rod drive mechanisms.
- The large coolant inventory in the primary results in large thermal inertia and long response time in the event of transients or accidents.
- Shielding requirements are reduced by the elimination of gamma sources of dispersed primary piping and parts.
- The large water volume between the core and the wall leads to a very low fast neutron dose over the RPV wall.
- Eliminating primary pumps and pressuriser results in lower costs, added safety, and advantages for maintenance and availability.

3. PLANT DESIGN

The CAREM nuclear island is placed inside a containment system, which includes a pressure suppression feature to contain the energy of the reactor and cooling systems, and to prevent a significant fission product release in the event of accidents.

The building surrounding the containment has been designed in several levels and it is placed in a single reinforced concrete foundation mat. It supports all the structures with the same seismic classification, allowing the integration of the RPV, the safety and reactor auxiliary systems, the spent fuels pool and other related systems in one block. The plant building is divided in three main areas: control module, nuclear module and turbine module.

Finally, CAREM NPP has a standard steam cycle of simple design.

CONCLUSIONS

The CAREM project consists of the development, design and construction of the prototype of an advanced small nuclear power plant. CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the reactor and also contribute to a higher level of safety. Some of the high level design characteristics of the plant are: integrated primary cooling system, self-pressurised, primary cooling by natural circulation, safety systems relying on passive features. Therefore, many technical and economical advantages are obtained with the CAREM design compared to conventional designs.

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CAREM PROJECT DEVELOPMENT ACTIVITIES

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Abstract

The CAREM project involves technological and engineering solutions, as well as several innovative design features that must be properly demonstrated during the design phase. Also specific codes used for modelling systems related with safety issues to obtain design parameters (e.g. primary cooling system, reactor core, fuel design, etc.) must be verified and validated against world-wide benchmark and/or experimental data to build confidence in their results. This paper describes main issues of the development program ongoing as part of the design phase of the CAREM project, which includes the design and construction of several experimental facilities and engineering mock ups. Main results obtained from the test facilities and validation of codes are also presented.

1. INTRODUCTION

The Argentinean CAREM project [1], which is jointly developed by CNEA and INVAP, consists of the development, design and construction of an advanced, simple and small Nuclear Power Plant (NPP).

CNEA has extensive experience in basic research and nuclear related technology in several areas. This includes nuclear fuel cycle (from uranium mining and U-enrichment to fuel manufacturing), waste management and disposal, production and uses of radioisotopes, food irradiation, technology of nuclear materials, nuclear I&C, operation and maintenance of research reactors, nuclear and radiological safety, etc.

INVAP has been involved as the designer and constructor of a wide range of technological projects. This includes research reactors, radioisotopes facilities, uranium enrichment, zirconium and beryllium processing, nuclear medical equipment, industrial waste treatment and disposal, environmental engineering and satellite construction among others.

So, the idea of design cycles has been applied in different frameworks involving several steps from the conceptual design to the final product (system, equipment, design code or technology process) capable of meeting the specific requirements. From early stages of the CAREM project, engineering has been underway in a globally planned sequence as part of this design cycle, where two general stages may be recognised:

(a) Conceptual/basic design, and experimental activities as an aid to design.

(b) Detail design, and experimental activities for validation/qualification.

Within the CAREM Project, the effort has been focused mainly on the nuclear island (inside containment and safety systems) where several innovative design solutions require developments of the **first stage** (to assure they comply with functional requirements). This comprises mainly: the **Reactor Core Cooling System (RCCS), the Reactor Core and Fuel Assembly, internals of the Reactor Pressure Vessel (RPV), and the First Shutdown System (FSS).** An extensive experimental plan has been prepared, including the design and construction of several experimental facilities to fulfil the Project's requirements.

An effort is planned for the systems/devices that require developments limited to the **2nd stage** of a Design Cycle (qualification, or need of adaptation of a proven solution); i.e. they are not actually innovative by their features, but require development effort in order to fit in the Project Engineering.

The **RCCS** modelling and qualification are boosted by the tests performed in a High Pressure Natural Circulation Rig (CAPCN), covering Thermal Hydraulics (**TH**), reactor control and operating techniques. The CAPCN rig reproduces all the dynamics phenomena of the RCCS, except for 3-D effects.

The Core Design involves different aspects i.e. study of thermal limits, neutronic modelling, structural mechanical and fuel assembly design. Neutronic modelling needs may be covered by benchmark data available world-wide and by experimental data from the Critical Facility RA-8. As for Fuel Assembly Design, CNEA has vast experience in the technology of nuclear fuels and structural and hydrodynamic tests will be carried out in Low and High pressures rigs.

The mechanical design of the core (structural, dynamic, seismic, etc.) and other **RPVI,** mock-up facilities are being constructed. They represent sections of the core, and include one vertical full-scale model with supporting Barrel and its Kinematics Chain.

The FSS, or more specifically the Control Rod Drives (CRD), is a good example of an innovative device design, comprising all the design cycle stages. An experimental plan is underway for the design and qualification stages.

The following is a brief description of some of the most relevant development tasks and their facilities that are carried out or foreseen as part of the CAREM project:

2. DYNAMIC TESTS OF RCCS

The purpose of the High Pressure Natural Circulation Rig: CAPCN (figure 1), is mainly to study the thermal-hydraulic dynamic response of CAREM primary loop, including all *the coupled phenomena that may be described by one-dimensional models*. This includes the validation of the calculation codes on models of the rig, and the extension of validated models to the analysis of the CAREM reactor. The main tool used in thermal-hydraulic calculations is RETRAN-02.

The CAPCN rig (see scheme figure 2) resembles CAREM in the primary loop (self-pressurised natural circulation) and the steam generator (helical once-through), while the secondary loop is designed only to produce adequate boundary conditions. Operational parameters are reproduced for intensive magnitudes (pressure, temperature, void fraction, heat flux, etc.) and scaled for extensive magnitudes (flow, heating power, cross-sections, etc.). The height was kept approximately on a 1:1 scale.

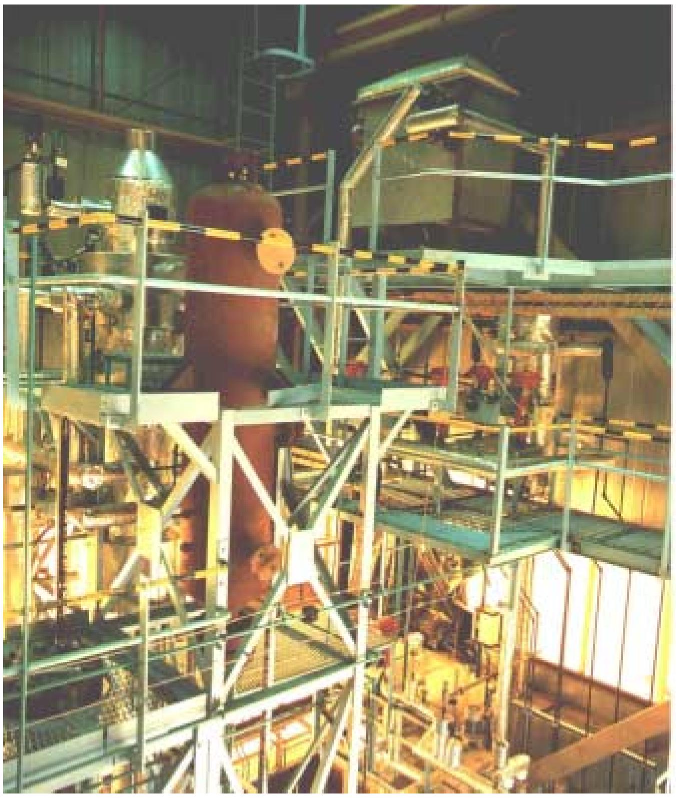

FIG. 1: CAPCN General View.

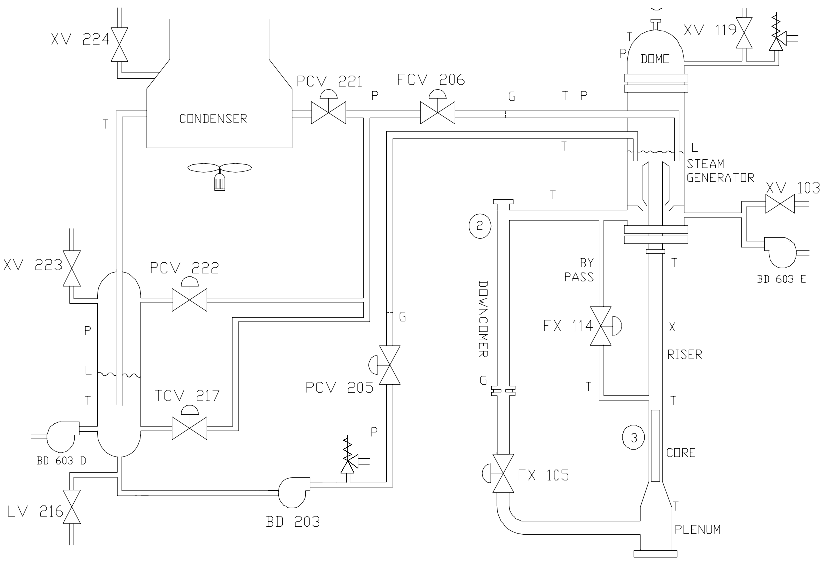

FIG. 2: CAPCN Simplified Process and Instrumentation Diagram.

The heating power may be regulated up to 300 kW, by the operator or by a feedback loop on primary pressure as a plain PID. An alternate feedback loop simulating core (neutronic) dynamics is under development.

The secondary loop pressure and cold leg temperatures are controlled through valves. The pump regulates the flow. The condenser is an air-cooled type with airflow control.
The control of the actuators (heaters, valves, pumps, etc.), data acquisition and operating follow up are carried out from a control room, through a PC based, multi-node software (flexible enough to define any feedback loop).

Most of the test [2] consists of an initial self-steady state in which a pulse-wise perturbation induces a transient. In this case the perturbation is a thermal imbalance as severe as possible for operational transients: thermal power is increased 12 kW (about 5% of FP) during 150 seconds. Primary pressure and circulating flow evolve mildly, with increases below 2 and 3% respectively, and primary temperatures hardly notice the perturbation. Therefore steam generation remains quite stable during the whole transient, a remarkable feature for a Steam Supply System (figure 3).

3. CHF TESTS AND THERMAL LIMITS

The TH design of the CAREM reactor core was carried out using an improved version of 3-D, two fluid model THERMIT code. In order to take into account the strong coupling of the thermal-hydraulic and neutronics of the core, THERMIT was linked with the neutronic code CITVAP. This coupled model allows the “drawing” of a 3-D map of power and thermal-hydraulic parameters at any stage of the burn-up cycle.

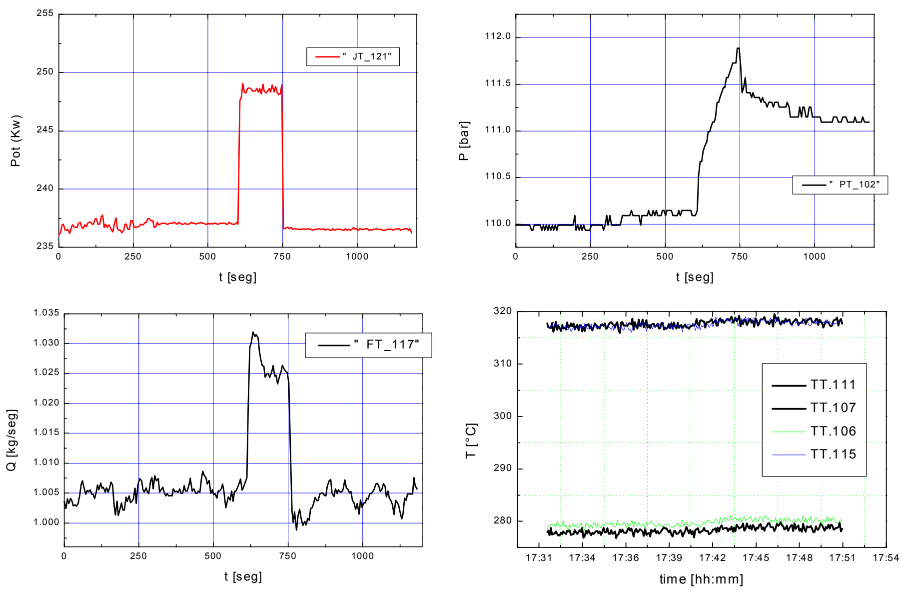

FIG. 3: Transient 1: +12 kW perturbation during 150 seconds.

The prediction of the thermal-limits (to harmful phenomenon like critical heat flux) of the fuel elements during operation and transients is considered of the utmost importance. The mass flow rate in the core of the CAREM reactor is rather low compared to typical light water reactors and therefore correlation or experimental data available are not completely reliable in the range of interest. Thus analytical data must be verified by ad-hoc experiments.

The experiments were conducted at the thermal-hydraulic laboratories of the Institute of Physics and Power Engineering (IPPE, Obninsk, Russian Federation). Figure 4 and 5 shows a view and scheme of the test section.

The main goal of the experimental program [3] was to generate a substantial database to develop a prediction methodology for CHF applicable to the CAREM core, covering a wide range of T-H parameters around the point of normal operation, i.e.:

Pressure	10 – 13	MPa
Mass Flux	200 –700	$kg/m^2/seg$
Quality	> -0.10	

FIG. 4: Test Channel with the 19 rod bundle simulator.

Most tests were performed using a low-pressure Freon rig, and results extrapolated to water conditions through scaling models. Finally a reduced set of tests were performed in water at high pressure and temperature, to validate the method for scaling.

Different test sections were assembled to simulate different regions in the fuel element as well as radial uniform and non-uniform power generations. A bundle with 35 % of the full length was tested to obtain CHF data under average sub-cooled conditions. More than 250 experimental points under different conditions were obtained in the Freon loop and more than 25 points in the water loop.

The preliminary analysis of results from Freon loop measurements show that some existing correlations present quite a good agreement.

4. FUEL ASSEMBLIES

The developments tasks on this subject comprise mainly the two following issues:

- The improvement and extension of the simulation models of the BACO computational code, that may be considered under stage one of the Design Cycle.
- The verification, evaluation and qualification of the designs, as a development under stage two of the Design Cycle.

The BACO code [4] produces a best-estimate computer simulation of the principal thermal-mechanical phenomena that occur within a nuclear fuel rod during burn-up process. It includes fission products generation and migration, fission gases release, in-clad pressure build-up, pellet deformation, crystallographic grain growth, stresses evaluation, pellet-clad interaction, etc.

This code had already been developed and verified against data of Fuel Assemblies of PHWR produced in Argentina. In order to cover Uranium Enriched Fuel Assemblies some new models had to be introduced and others had to be modified. These include the influence of high burn-up on thermal conductivity of UO_2, the thermal conductivity in the pellet-clad gap (influence of Xe at high burn-up) and the migration of porosity (densification and restructuring).

These new models were validated through the participation in a Co-ordinated Research Project (CRP) of the International Atomic Energy Agency [5]. This CRP, called FUMEX Program, produces validation by initially sharing the experimental information of operating conditions and requirements of a certain fuel, and comparing "blind simulation" results with experimental measurements.

The BACO code, combined with International Fuel Performance Experiments Database (of the OECD Nuclear Energy Agency) should cover the validation and evaluation requirements of the fuel rod design.

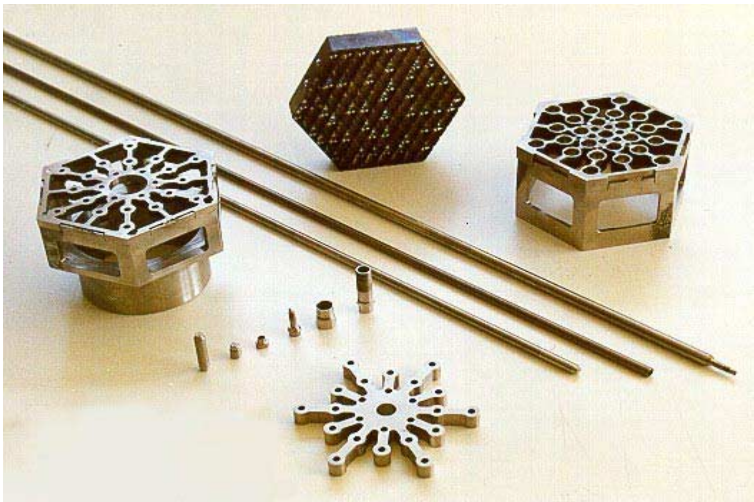

FIG. 5: Main fuel elements components during development stage.

The fuel assemblies and absorbing clusters (Figure 5) will be subject to a series of qualification tests, including standard mechanical evaluations, and hydraulic tests. The latter comprise:

- Tests in a Low Pressure Rig evaluating pressure-losses, flow-induced vibrations and general assembling behaviour.
- Endurance tests in a high pressure loop points to wear-out and fretting issues.

5. NEUTRONIC MODELLING VALIDATION

Validations against VVER reactor geometry [6, 7, 8] were made using experimental data from the ZR-6 Research Reactor, Central Research Institute for Physics, Academy of Sciences, Hungary, A series of benchmark data were used for typical PWR reactor. Further experimental data will be obtained from Critical Facility RA-8, which resembles certain CAREM neutronics issues. The neutronic calculation line used for the CAREM reactor core, the nuclear data library, and the validations that have already been made, are presented.

Nuclear Data: The HELIOS library (190, 89 and 34 groups) used by the CONDOR code, has been especially developed to a group-wise (not an isotope-wise) order. Almost all the data in the library are based on the ENDF/B-VI data files.

Cell Code CONDOR 1.3: This code has the capability to calculate nuclear fuel elements with its spatial detail (without homogenisation). Collision Probabilities method (CPM) is used in a general 2-D cylindrical geometry. The Carlvik method with the macroband algorithm is applied to obtain the CP by the double numerical integration. The program uses normalisation schemes on integration chords (it preserves the regions volumes and surfaces) and CP (it preserves the reciprocity and balance between them).

The code CONDOR has been validated against VVER and PWR critical experiments in different conditions (fuel, moderator, H_2O/U ratio, etc.). The validation against 26 VVER-type cells gave the result Keff=0.9953 ± 0.0089. Also the validation against 110 cases of UO_2 systems gave Keff = 1.00065 ± 0.0080 and over 86 cases of U-metal systems gave Keff=1.0044 ± 0.0043.

Core Code CITVAP: This is a code derived from the well known diffusion code CITATION II, adding the following options using macroscopic cross-sections: Burnup, Fuel Management and Positioning of control rods. Preserving the entire original CITATION options, the most relevant features implemented in CITVAP are the following:

- Operation follow-up capabilities (fuel management and rod movements).
- Greater versatility in the input data improving 3 D problem descriptions and makes it easier to describe fuel management and fuel element movements.

The code has been validated against VVER cell type reactors giving Keff=0.997±0.003 and against MTR reactors giving Keff=1.004±0.006.

6. HYDRAULIC CRD TESTS

One of the most innovative systems behind the CAREM concept is the Hydraulic (in-vessel) Control Rod Drive mechanism HCRDM. Two designs are under development: “Fast Extinction” and “Adjust & Control” CRD, being the latter that presents major challenges related to the design (Figure 6.).

The design embraces mechanical and thermal-hydraulic innovative solutions so feasibility of the concept must be demonstrated as a first step to be included in the reactor engineering.

In the other hand, their operational functions (the Adjust and Control, and the Fast Extinction) are part of one of the most important safety systems of the reactor: the First Shutdown System

(FSS). These two features mean that a complete experimental program including "experiment-aided design" and qualification tests, is necessary to achieve the high reliability performance jointly with low maintenance.

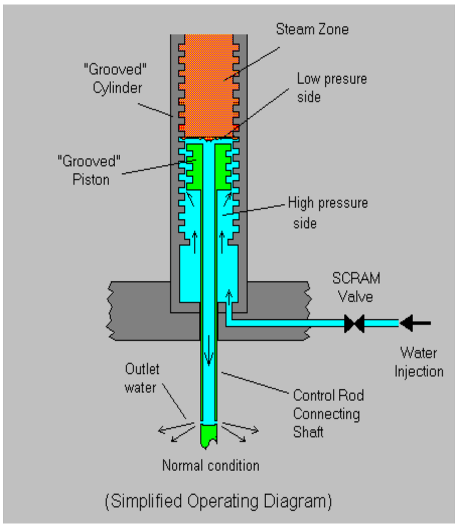

FIG. 6: HCRDM Adjust and Control System.

The development plan refers to four well separate stages and includes the construction of several experimental facilities, reaching the testing of the system performance under RPV operating conditions. The four different stages and their (built or foreseen) facilities are:

6.1. Preliminary tests (conceptual verification)

The aim of this test was to prove the feasibility of the theoretical approach, to have a first idea of some of the most sensitive controlling parameters and to determine spot points to be focused during design. Tests were undertaken on a rough device with promising experimental results, and good agreement with first modelling data was obtained.

6.2. First prototype tests

This stage pointed to determining preliminary operating parameters on a full-scale mechanism as a first approach towards detail engineering. These parameters include range of flow, ways to produce hydraulic pulses, etc. Manufacturing hints that simplified and reduce costs of the first design were also found. Tests were carried out in a craftily built rig and as part of this experimental development it was decided to separate the regulating and fast-drop requirements in different devices.

6.3. Test on a low pressure loop

This stage was carried out with the CRD at atmospheric pressure, and with feed-water temperature regulation up to low sub-cooling. The feed-water pipeline simulated alternative configurations of the piping layout with a second injection line (dummy) to test possible interference of pulses.

The ad-hoc test loop (CEM, Circuito de Ensayo de Mecanismos, figure 7) was designed to allow automatic control of flow, pressure and temperature, and its instrumentation produces information of operating parameters including pulse shape and timing. The tests included the characterisation of the mechanism and the driving water circuit at different operating conditions, and the study of abnormal situations as increase in drag forces, pump failure, loss of control on water flow or temperature, saturated water injection, suspended particle influence, and pressure “noise” in feeding line.

The tests carried out in the turbulent regime, which are the closest conditions to operation obtained in this loop, showed good reliability and repetitiveness as well as sensitivity margins for the relevant variables within control capabilities of a standard system.

FIG. 7: Low Pressure Mechanism Test Rig.

6.4. Qualification Tests

A high-pressure loop (CAPEM, Circuito de Alta Presión para Ensayo de Mecanismos) is being designed in order to reach the actual operating conditions (P = 12.25Mpa, T ≈ 326°C). The main objectives are to verify the behaviour of the mechanisms, to tune up the final controlling parameter values and to perform endurance tests. After this stage, the system under abnormal conditions, such as the behaviour during RPV depressurisation, simulated breakage of feeding pipes, etc. will be tested.

7. RPV INTERNALS TESTS.

The mechanical structure of the core, supporting guides and all parts of the kinematic chain of the First Shutdown System are of particular interest. Complex assemblies and structures like the Steam Generator Units or ad-hoc mechanical solutions require the evaluation of manufacturing and assembly process, before finishing the design stage.

In sum, internals must be verified in order to define manufacturing, assembling allowances, and other detailed engineering parameters to comply with their function during the RPV lifetime. Most tests are performed on mock-up facilities at 1:1 vertical scale. The following is a brief description of some of these devices, experimental plans and current status.

7.1. Full Scale Core-Sector

This is a complete-vertical representation of the core up to an extension of three fuel elements (i.e. structure, upper and lower grids, dummy FE, absorbing element guides, etc.) and major devices involved (i.e. absorbing fuel rods and connecting bar). All the structures can be perturbed by a hydraulic-driven actuator, which simulates minor vibrations and horizontal seismic loads on a wide range of frequencies and magnitudes (figure 8).

The aim is to make fine adjustments to the design, to verify couplings and auxiliary tools and to give a glance at the vibration modes of the whole assembly.

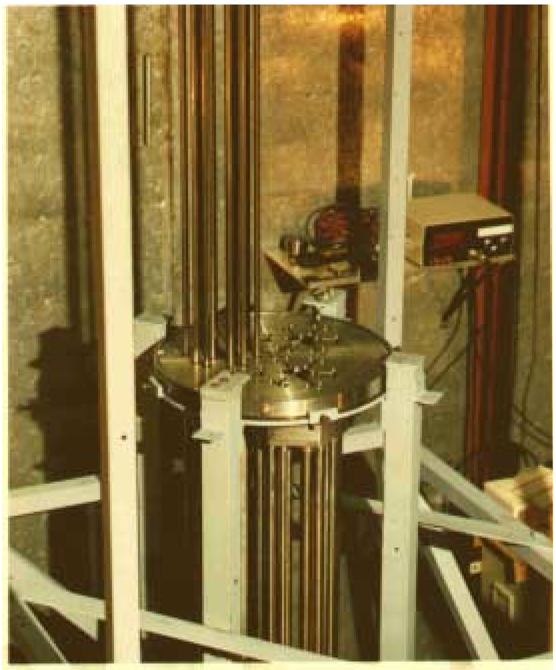

Figure 8: Full Scale Core-Sector.

First series of experiments, have already been completed with encouraging results: the insertion of absorber rods, both in stepwise movement and rapid fall, was not affected by the perturbation of the whole dummy in a broad range of frequencies.

Also improvements in the guide devices for absorbing elements showed better performance while reducing manufacturing complexity from previous design. A second series of experiments to be performed under water are already planned.

7.2. Full Scale (Vertical) Structural Barrel, Core and Cinematic Chain:

An important series of experiments to verify structural and dynamic behaviour of the Barrel will be performed after finishing those at the core sector. Being this structure very slender, the experiments deal with alignment, clearances in linear bearings, mass momentum and dynamic analysis. The latter point to determine natural frequencies, mode shapes and responses of the system under various external perturbations, which resemble seismic conditions and other vibrations.

The facility contains a complete sector representation (up to three fuel elements), similar to the Full Scale Core-Sector, including a sector of the Barrel and the Bar Guide Column (BGC) and one complete CRD System (hydraulic mechanism and all related parts of the cinematic chain including the absorbing element).

The whole device will be used to verify the performance of the Fist Shutdown System under normal and abnormal circumstances (i.e. misalignments or seismic) measuring the total elapsed time for rod dropping after SCRAM triggering. The vertical full scale will help to evaluate refuelling and maintenance manoeuvres, which have to be done underwater and far from the core region.

8. IN VESSEL INSTRUMENTATION

Since the HCRD design adopted has no movable parts outside the RPV, it was necessary to design a special probe to measure the rod position able to withstand primary environmental conditions. The proposed design consists in a coil wired around the HCRD cylinder with an external associated circuit that measure electric reluctance variations induced by the movement of the piston-shaft (made of magnetic steel) inside the cylinder.

Cold test performed showed that the system is capable of sensing one step movement of the regulating CRD, with an acceptable accuracy. In-furnace high temperature tests are going to be conducted to evaluate the behavior of the system against temperature changes similar to those occurring during operational transients.

The design of special high pressure removable feedthrouhs to allow dozens of electrical signals passing through RPV cover is also under work. This means the development and qualification of specific manufacturing and welding techniques.

CONCLUSIONS

The development program of the CAREM project has been driven to demonstrate the robustness of the design as well as being an irreplaceable tool to help the designers in issues related with the effectiveness and reliability of systems and components important to safety.

Relevant progress has been obtained for the validation of thermal-hydraulic, neutronics and fuel codes using benchmark and experimental data.

Encouraging experimental results have been obtained towards the validation of the in-vessel hydraulic CRD design and the construction of a high pressure facility to perform tests at operating conditions is foreseeing.

Future work on development activities will be mainly focused on the qualification of safety systems as well as the manufacturing process for non-commercial or non-well qualified components, as those required by licensing program.

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IAEA-SR-218/28

ADVANCED DESIGN FEATURES ADOPTED IN SMART

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Abstract

SMART is a promising advanced small and medium category nuclear power reactor. It is an integral type reactor with a sensible mixture of new innovative design features and proven technologies aimed at achieving highly enhanced safety and improved economics. The enhancement of safety and reliability is realized by incorporating inherent safety improving features and reliable passive safety systems. The improvement in the economics is achieved through system simplification, component modularization, construction time reduction, and increased plant availability. Preliminary safety analyses on selected limiting accidents confirm that the inherent safety improving design characteristics and the safety systems of SMART ensure reactor safety.

1. INTRODUCTION

The trend in the current development of nuclear reactors is towards enhanced safety and reliability, noticeable economic improvement, and reduction of radioactive waste production compared to existing reactors. The advanced technology developed during the past few decades and accumulated experiences in reactor operation provide firm foundations for the achievement of these requirements. It is widely acknowledged that the realization of such high-level requirements is possible only with small and medium sized reactors (SMR).

Various advanced types of SMR are currently under development worldwide, and some of them are ready for construction. One beneficial advantage of SMR is the easy implementation of advanced design concepts and technology. Drastic safety enhancement can be achieved by adopting inherent safety characteristics and passive safety features. Economic improvement is pursued through system simplification, modularization, and reduction in the construction time.

SMART (System-integrated Modular Advanced ReacTor), a small sized integral PWR with the rated thermal power of 330MW [1] is one of those advanced types of SMR. It is currently under basic design at KAERI (Korea Atomic Energy Research Institute). In the design of the SMART, safety enhancement and economic improvement are the most important considerations. Further, drastic reduction of liquid waste production is one important design philosophy. Design features contributing to safety enhancement are basically inherent safety improving features and passive engineered safety features. Features improving the economics are ones such as system simplification, component modularization, on-shop fabrication & site installation, and other features reducing the construction time.

This paper presents various advanced design features adopted in SMART with respect to safety enhancement and economic improvement. The results of preliminary safety analyses on selected design basis events are also briefly presented.

2. ADVANCED DESIGN FEATURES

2.1. Inherent Safety Improving Core Design Features

The design of the reactor core of SMART is based upon existing PWR technology and the fuel designs utilized in currently operating power reactors in Korea. Improved inherent safety results from the nuclear design characteristics of the SMART core with the adoption of advanced design features specifically such as low power density and no soluble boron operation.

During normal power operation, the average core power density of SMART is only about 60% compared to currently operating power reactors with the same fuel design. This feature of low core power density design provides ample core thermal margins with regard to the critical heat flux to accommodate any power transients and thus to ensure the core thermal reliability during power operations. At normal design conditions, the analysis shows the core thermal margin is over 15%. The analysis further shows that the core progresses to a stable state with respect to power distribution due to the enhanced negative temperature feedback effect.

The soluble boron-free core design inherently produces a strong negative moderator temperature coefficient over the entire cycle. Analyses indicate that this design provides much improved response to a variety of system transients and load changes in terms of reactor control and safety. The core reactivity during the normal operation is controlled by solid burnable absorbers and through fine movement of the control rods. The soluble boron-free design together with the low core power density inherently ensures the SMART core to be more stable and resistant to transients, and hence provides improved operational flexibility. The elimination of soluble boron from the primary system further provides various benefits and contributes to the economic improvement of the reactor system. Those benefits will be described in section 3.

The axial zoning of the burnable poisons (BP) is another advanced design feature of the SMART core. The use of the control rods during normal operation to control the excess reactivity causes the power distribution to skew towards the lower part of the core. Thus, solid BP rods having axially different effective lengths are adopted in the SMART core to adjust the skewed power distribution.

TABLE I. DESIGN CHARACTERISTICS OF THE SMART CORE

Fuel type	17 × 17 UO_2 square FA	Excess reactivity (hot full power, eq. Xe, %Δρ)	2.96 (BOC), 2.40 (MOC) 1.41 (EOC)
Active fuel length (m)	2.0	Shutdown margin (cold zero power, %Δρ)	3.16 (BOC), 2.90 (MOC) 4.11(EOC)
Enrichment (w/o)	4.95	Maximum peaking factor (hot full power)	2.35
No. of fuel assemblies (FA)	57	Mod. temp. coeff. (hot full power, pcm/°C)	-72 < MTC < -42
Core power density (w/cc)	62.6	Fuel temp. coeff. (hot full power, pcm/°C)	-4.52 < FTC < -2.54
Refueling cycle (yr)	> 3	Scram worth (pcm)	29707(BOC) 28785(MOC) 31018(EOC)
No. of control element banks	49		
Control element material	Ag-In-Cd		
No. of BP rods per FA	24 - 28		
Burnable poison materials	Al_2O_3-B_4C, Gd_2O_3-UO_2		

The SMART core region is shrouded with shielding materials of several sheets of stainless steel plates on the side and bottom. These plates are located outside of the core barrel and the gaps between plates are filled with water. This shielding greatly reduces the fast neutron fluence on the reactor pressure vessel(RPV) and thus maintains the high-level integrity of the RPV. The fast neutron fluence is calculated to be around 2×10^{16} n/cm^2 at the end of the RPV lifetime of 60 years that is much lower compared to currently operating loop-type conventional reactors.

The design characteristics with the key design parameters of the SMART core are summarized in Table 1.

2.2. Advanced Features of NSSS Components

2.2.1 Reactor Vessel Assembly (RVA)

The prominent design feature of SMART is the adoption of an integral arrangement. All the primary components such as core, steam generators, main coolant pumps, and pressurizer are integrated into a single pressurized vessel without any pipe connections between those components. Figure 1 shows the structural configuration of the SMART reactor assembly. Four(4) main coolant pumps are installed vertically at the top of the reactor pressure vessel(RPV). The reactor coolant flows upward through the core and enters into the shell side of the steam generator(SG) from the top of the SG. The SGs are located at the circumferential periphery between the core support barrel and RPV upper the core.

The large volume at the top part of the RPV is used as a self-pressurizer. This integral arrangement of the major components into a single RPV causes a difference in the design concepts of the RVA compared to the conventional loop type reactors. While the overall arrangement of the reactor coolant system (RCS) is simplified by the elimination of the primary piping systems, the layout within the RPV becomes more complex. As the only single pressure boundary is devised to encompass all the primary components, the global behavior of the system will be highly dependent upon the mutual interactions caused by each component. Design basis dynamic events such as seismic events and failure of piping are considered during the design process.

2.2.2 Steam Generator (SG)

Twelve identical SG cassettes are located in the annulus formed by the RPV and the core support barrel. Each SG cassette is of once-through design with helically coiled tubes wound around the inner shell. The primary coolant flows downward in the shell side of the SG tubes, while the secondary feed-water flows upward in the tube side. Therefore, the tubes are under compressive loads from the greater primary pressure, reducing the stress corrosion cracking and thus reducing the probability of tube rupture. The steam exits the SG with 40°C superheated steam condition at normal operation and thus a steam separator is not required. Three(3) steam and feedwater pipes from the adjacent steam generator cassettes are connected together to form a section. There are thus a total of four(4) sections in SMART. If there is a

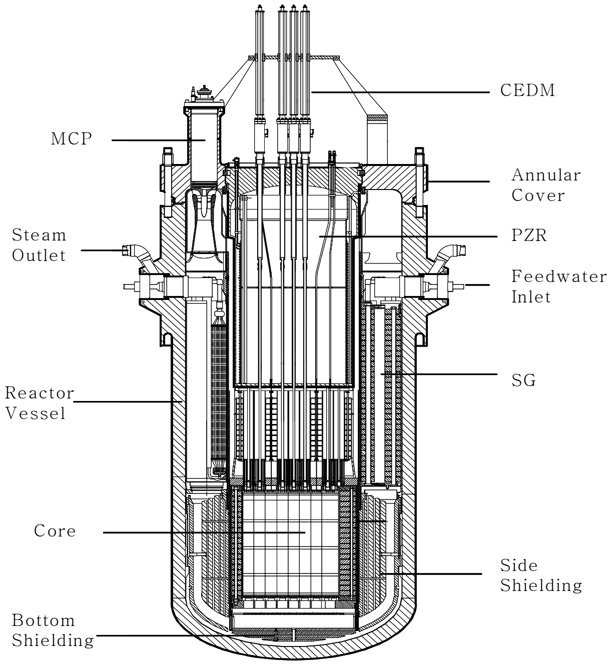

Figure 1. The SMART Reactor Assembly.

leakage in one or more of the tubes, the relevant section is isolated and SMART can be operated with reduced power until the scheduled shutdown. With the adoption of a modular concept, any defective SG can be replaced individually.

2.2.3 Pressurizer (PZR)

The SMART PZR is an in-vessel self-controlled pressurizer located in the upper space of the reactor assembly and is filled with water, steam and nitrogen gas. The self-pressurizing design eliminates the active mechanisms such as spray and heater. The pressure is controlled only by the partial pressure of nitrogen gas. To enhance the self-pressurizing capability, the PZR operates at temperature of 100°C and at pressure of 15MPa. Thus, a PZR cooler is installed to maintain the low PZR temperature, and wet thermal insulator is installed to reduce the heat transfer from the primary coolant. The large pressure variation during normal operation can be reduced by keeping the average primary coolant temperature constant with respect to power change.

2.2.4 Control Element Drive Mechanism (CEDM)

Due to the soluble boron-free operation, an important design requirement for the SMART CEDM is a fine maneuvering capability to control the excess core reactivity. The conventional magnetic jack-type CEDM is not able to meet this requirement. Thus, a linear step motor type CEDM with easy access for in-service inspection and replacement is employed. The minimum step length is 4mm per step that is short enough for the fine reactivity control. Forty-nine(49) CEDMs are installed in the fifty-seven(57) fuel assemblies of the SMART core. There may exist electro-magnetic interference between CEDMs due to crowded installation on the limited space of the RPV head, and thus accidental mal-function may be induced. However, the magnetic-field effect analysis shows that the maximum field density is about 4.7×10^{-6} Tesla that is far below the required field density, 1.5×10^{-3} Tesla for the position indicator operation, and thus no electro-magnetic interference effect between CEDMs exists.

2.2.5 Main Coolant Pump (MCP)

The SMART MCP is a canned motor type pump that eliminates the problems of conventional seals and associated systems. In other words, the canned motor type pump basically eliminates the small break loss of coolant accident (SBLOCA) associated with a pump seal failure. Four(4) MCPs are installed vertically on the RPV annular cover. MCP is an integral unit consisting of a canned asynchronous 3-phase motor and an axial flow single-stage pump. The motor and pump are connected through a common shaft rotating on three radial and one axial thrust bearings. The impeller draws the coolant from above and discharges downward directly to the SG. This design minimizes the pressure loss of the flow.

2.2.6 Advanced Man-Machine Interface (MMI) Technology

SMART adopts advanced instrumentation & control (I&C) and MMI technology to enhance the reliability and safety of the reactor system by greatly improving the control capability. Digital technologies and equipments are employed in the protection and control system. The advanced IMS (Instrumentation and Monitoring System) ensures the achievement of high reliability and performance of the system with several advanced features such as digital technology, remote multiplexing technology, and signal validation and fault diagnostics technology. The main control room is designed with in-depth application of MMI technology. The reactor is operated and controlled by a single operator.

Table 2 summarizes some of key design parameters of major components of the SMART system.

TABLE 2. KEY DESIGN PARAMETERS OF SMART MAJOR COMPONENTS

Reactor vessel assembly (RVA)		Operating temperature (°C)	100
Overall length (m)	10.7		
Outer diameter (m)	5.05	*Control element drive mechanism*	
Average RV thickness (mm)	256	*(CEDM)*	Linear pulse motor
Vessel material	SA508, CL-3	Type	driven
			49
Reactor coolant system (RCS)		No. of CEDM	4.0
Design pressure (MPa)	17	Step length per pulse (mm)	
Design temperature (°C)	350		
Operating pressure (MPa)	15	*Main coolant pump (MCP)*	
Core inlet temperature (°C)	270	Type	Canned motor axial
Core outlet temperature (°C)	310		pump
Total flow rate (kg/sec)	1,550	No. of MCP	4
		Flow rate (m^3/hr)	2,006
Steam generator (SG)		Water head (m)	17.5
Type	Once-through		
	helical tubes	*2.2.6.1 Secondary system*	
No. of SG cassettes	12	Feed-water temperature (°C)	180
Tube material	Ti-alloy	Feed-water pressure (MPa)	5.2
		Steam temperature (°C)	274
Pressurizer (PZR)		Steam pressure (MPa)	3.0
Type	Self controlled	Degree of superheating (°C) at	40
	Gas PZR	normal operation	

2.3. Passive Engineered Safety Features

Besides the inherent safety improving design features implemented into SMART, further safety enhancement is accomplished with highly reliable and advanced engineered safety systems. These systems are designed to function passively on demand. The following is a summary of major safety systems adopted in the SMART design. Figure 2 shows the schematic configuration of SMART Nuclear Steam Supply System (NSSS) including engineered safety features.

2.3.1 Reactor Shutdown System (RSS)

The shutdown of SMART is achieved by one of two independent systems. The primary shutdown system are control rods containing Ag-In-Cd absorbing material. On demand at the normal case, the shutdown signal de-energizes the CEDM and then the control rods drop into the core by gravity. The drop time is approximately 8 seconds. In the case of the failure the primary shutdown system, the emergency boron injection system is provided as an active back-up. One of the two trains is sufficient to bring the reactor to sub-critical condition.

2.3.2 Passive Residual Heat Removal System (PRHRS)

The PRHRS removes the core decay heat by natural circulation in emergency situations. SMART has four(4) independent PRHRS trains with 50% capacity for each train in core decay heat removal, and the operation of two trains is sufficient to remove the decay heat. The system provides the long-term cooling and is capable of decay heat removal for a minimum of 72 hours without any corrective action by operators for the design basis accidents.

2.3.3 Emergency Core Cooling System (ECCS)

The core during a SBLOCA is protected and covered with by the large primary coolant inventory and the pressure balance effect between the primary system and the safeguard vessel surrounding the RPV. When the primary system pressure drops below 10 MPa, the valve in the line of the ECCS is automatically opened and the water is injected immediately into the core by gas pressure. The ECCS consists of two (2) independent trains with 100% capacity for each train. The system provides vessel refilling so that decay heat removal system can function properly in the long-term recovery mode following the event.

2.3.4 Reactor Over-Pressure Protection System (ROPS)

The function of ROPS is to reduce the reactor pressure under the over-pressurizing design basis accidents. The system consists of two(2) parallel trains connecting the PZR and the external shielding tank through a single pipeline. When the primary system pressure increases above 17MPa, pilot operated safety relief valve (POSRV) on both trains are opened automatically to discharge the steam into the external shielding tank.

2.3.5 Safeguard Vessel (SV)

The SV is a leak-tight pressure-retaining steel vessel which accommodates all primary reactor systems inside. The pressure in the SV is kept at 3MPa. The primary function of the SV is to confine the radioactive products within the vessel and thus to protect any primary coolant leakage to the containment. The vessel also functions to keep the reactor core undamaged during the postulated design basis accidents including SBLOCA, with the operation of the PRHRS and ECCS. The steam released into the SV at the postulated design basis accidents is sparged into the external shielding tank by the passive opening of the SV relief valve.

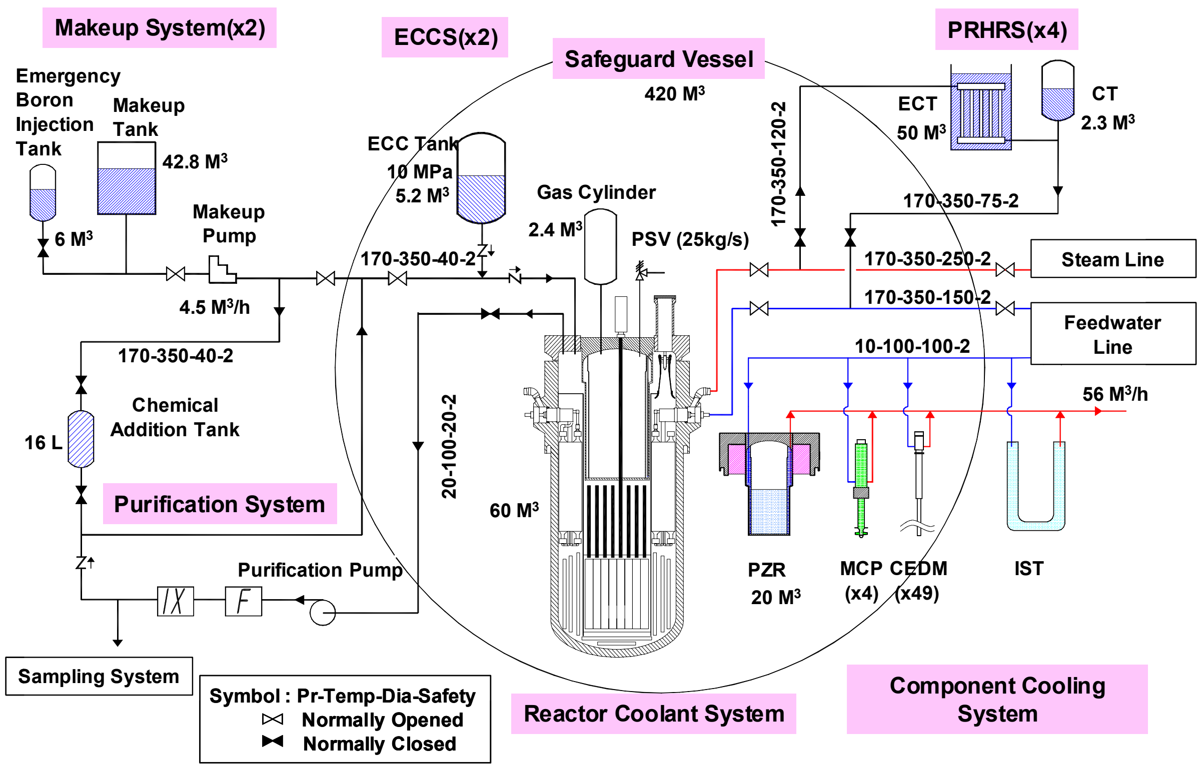

FIG. 2. The Schematic Configuration of the SMART NSSS

3. ECONOMIC IMPROVING FEATURES

The major economic improving features for SMART can be summarized as system simplification, component modularization, factory fabrication & direct site installation of components, reduced construction time.

The integral arrangement of the primary reactor systems requires only a single pressurized vessel and removes large-sized pipes connecting primary components. The adoption of the simplified passive systems provides a net reduction in the number of safety systems, and also drastically reduces the number of valves, pumps, wirings and cables, pipes, etc. The soluble boron-free design is one of most important design features that largely contribute to the system simplification by allowing the removal of associated systems and components required for boric acid processing, chemical volume and control systems. This feature also minimizes the liquid radwaste generation and thus simplifies the associated processing systems.

A simplified modular design approach is applied to all SMART primary components. Optimized and modularized small-sized components allow easy factory fabrication and direct installation at site, and thus lead to shorten the construction time and schedule. These features allow a construction period of less than three(3) years from first concrete to fuel load. The compact and integral primary system also eliminates the complexity and extra components associated with the conventional loop-type reactors.

SMART is designed for a sixty (60) year life and for a three (3) year fuel cycle with a single or one and half batch refueling scheme. The neutron fluence to the reactor vessel is greatly reduced by specially designed side and bottom shielding. The availability factor of the SMART plant is 95%, and the occurrence of unplanned automatic scram event is less than one per year.

There are some other features contributing to economic improvements. SMART uses advanced on-line digital monitoring and protection system that provides significant advantages of increasing the system availability and operational flexibility. The adoption of advanced man-machine interface technology leads to the reduction of human errors and to a compact and effective design of the control room with respect to minimizing the staff requirements.

4. SAFETY OF THE SMART REACTOR

The important design features directly contributing to the safety enhancement of the SMART reactor system are the inherent safety improving core design features and passive engineered safety systems. In addition, the adoption of the on-line digital monitoring and protection system and advanced man-machine interface technology contribute to enhanced system responses and thus system reliability. Currently, a full set of safety analyses with basic design data is in progress.

The integral arrangement of the primary system fundamentally eliminates a large break loss of coolant accident that is the most significant design basis accident of the conventional loop-type reactor. A number of design basis accidents that are primarily considered as limiting events were analyzed and briefly presented here [2]. The selected events are main steam line break (MSLB), main feed-water line break (MFLB), total loss of flow (TLOF), and a postulated accident of SBLOCA. The failure of one PRHRS train is considered as a single failure assumption for the analyses. The analysis was performed using MARS/SMR computer code[3] developed at Korea Atomic Energy Research Institute (KAERI) by consolidating and restructuring RELAP5/ MOD3.2.1.2 and COBRA-TF computer code. Various analysis models for the SMART specific designs are included in MARS/SMR.

The MSLB is a limiting accident for the decrease in heat removal by the secondary system, and may occur as a result of thermal stress or cracking in the main steam line. A double-ended guillotine type break of the steam pipe is assumed for this accident analysis. Only two trains of the PRHRS are assumed to be available for the system cooldown. For this accident, the hot channel departure from nuclear boiling ratio (DNBR) is a major parameter of concern. This accident causes the power to reach the high power trip set-point of 115% at 8 seconds due to the positive reactivity insertion and the reactor trip occurs at about 10 seconds after the event initiation. The increased core power then rapidly drops by the insertion of the control rods. The minimum DNBR of 1.34 is reached at 13 seconds, which is higher than the specified acceptable fuel design limit (SAFDL) DNBR of 1.3. The primary and secondary pressures are well below the safety criteria of 110% of the design pressure, 18.7 MPa. Based on the result, the core flow rate is slightly increased to ensure the higher DNBR at this accident.

The degradation of heat removal by the secondary system can occur by the limiting event of MFLB. A double-ended guillotine type break of the feedwater line between the feedwater isolation valve and the inlet of the SG is assumed. A rupture of the feedwater line results in a rapid increase in the primary pressure due to the reduced heat removal by the secondary system. The reactor and MCP trip occurs at about 6 seconds with the high pressure trip signal after the initiation of the accident. Upon a reactor trip, the PRHRS comes into operation and starts to remove the core decay heat. The initial core power increase due to the positive reactivity insertion causes the hot channel DNBR to decrease, and a minimum DNBR of 1.44 is reached at around 6 seconds. The analysis shows that the MFLB is bounded by the MSLB accident and thus becomes a less limiting event in SMART.

The TLOF is a typical design basis accident of decrease in the reactor coolant flow rate and is caused by a complete loss of power supply to all MCPs in operation. This accident results in a complete loss of forced circulation of primary coolant flow and thus causes the largest degradation in the DNBR margin. The only credible mechanism for a loss of power supply to all four MCPs is a loss of offsite power. Upon the loss of offsite power, turbine trip, feedwater flow termination, and coast down of all MCPs occur concurrently. A single failure of the PRHRS is assumed. The decrease in the coolant flow due to the coast down of MCPs causes an increase in the core average coolant temperature, which in turn causes a decrease in the core power by the negative MTC effect. The core power then drops rapidly to the decay heat level due to the insertion of the control rods. The hot channel DNBR decreases with a decrease in the core flow and an increase in the coolant temperature. The analysis shows that the DNBR reaches a minimum value of 1.53 at 6.5 seconds as the core power rapidly decreases. 150 seconds after the initiation of the accident, the natural circulation flow in the PRHRS loops reaches 5.7% of the initial feedwater flow, and the primary and secondary temperatures decrease at a rate of 90°C/s. The primary and secondary pressures are well below the safety criteria of 110% of the design pressure, 18.7 MPa.

For the SBLOCA analysis, the instantaneous guillotine rupture of the pipeline connecting the PZR end cavity and the N_2 gas cylinders is considered. In this accident, the primary coolant is released into the SV. The rupture of the gas cylinder pipe causes N_2 gas in the PZR end cavity to discharge through the break and thus the primary system pressure rapidly decreases. When the system pressure reaches down to the low-pressure trip setpoint of 12MPa, the reactor trip occurs at about 16 seconds after the initiation of the accident. The PRHRS comes into function and starts to remove the decay heat after the main feedwater isolation valves are closed by the reactor trip signal. The ECCS is actuated at around 80 seconds after the initiation of the accident when the primary system pressure decreases below 10MPa. The continuous steam discharge through the break causes the primary system inventory to decrease and the pressure of the SV to increase. The pressures of the primary system and SV equalize at around 9,300 seconds, and the break discharge flow ceases and no more system inventory loss occurs. The analysis shows that the collapsed water level is maintained at about 1m above the top of the core. It is thus shown that the proper function of the ECCS with three trains of PRHRS adequately removes the core decay heat and mitigates the consequences of the accident, and thus secures the reactor to a safe condition in the case of a postulated SBLOCA.

CONCLUSIONS

An advanced small-sized integral type reactor, SMART has been developed for dual-purpose applications for small-scale power generation and an energy source for nuclear seawater desalination. SMART is an innovative design to achieving a high degree of safety by adopting inherent safety improving features and passive safety systems. Economic competitiveness, despite its small sized capacity, is achieved through system simplification, higher availability, modularization, reduced construction time, etc. Preliminary safety analyses on the selected limiting events show that the SMART system properly functions and thus secures the reactor to a safe condition.

ACKNOWLEDGEMENT

The development of SMART has been carried out as one of national long-term nuclear R&D projects financially supported by the Ministry of Science and Technology of the Republic of Korea.

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600 MW(e) CLASS KSNP DESIGN

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Abstract

The small KSNP (Korean Standard Nuclear Power Plant) of 600 MW(e) is being developed by KOPEC (Korea Power Engineering Company, Inc.), with the goal of introducing a proven and licensable medium size reactor to the countries which may need such size reactor due to their lower national grid capacity or for other reasons. Its reference plant is the KSNP, one of the most advanced 1000 MW(e) class pressurized water reactors (PWR) currently operating in the world. Major components are reduced in size to incorporate the power reduction while the proven basic design concepts of the KSNP are maintained. Advanced design features and accumulated operating experiences are reflected into the system design. Once the design is completed and the licensability is proven, the small KSNP will be one of the most evolved and competitive 600 MW(e) class PWRs.

1. INTRODUCTION

It is generally agreed that reactors of larger size are preferred when the economy of scale is considered alone. If the cost of grid, however, is considered to maintain efficiency and reliability of the power plant, smaller reactors usually turn out to be more economical until the larger ones are justified later. In addition to such technical and economical concerns, smaller reactors are believed to help ease financing concerns regarding the initial investment.

Under these circumstances, KOPEC (Korea Power Engineering Company, Inc.) has begun to develop a medium size reactor of 600 MW(e), named the small KSNP (Korean Standard Nuclear Power Plant) [1]. The design goal is to introduce a proven and licensable medium size reactor to the countries which may need such size reactor due to their lower national grid capacity or for other reasons. The small KSNP is being designed after the 1000 MW(e) KSNP [2], the reference plant of the small KSNP. The designs are basically identical each other except for the difference in power output and some design improvement, and conform to the same licensing requirements, and codes and standards.

In this paper, the general design characteristics of the small KSNP, specifically in the nuclear steam supply system (NSSS) design, are described and compared with those of the 1000 MW(e) KSNP. Presented also are the preliminary quantitative accident evaluation results for the limiting design basis events.

2. DESIGN APPROACH AND IMPROVEMENTS

The basic design approach for the small KSNP is (1) to downsize the components of the 1000 MW(e) KSNP according to the power reduction ratio if applicable, (2) to maintain the proven design concept of the 1000 MW(e) KSNP, (3) to incorporate the up-to-date design improvement and operating experiences, and (4) to meet the current regulatory requirements.

Up-to-date design improvements and accumulated operating experiences are reflected in the system design. The unique or advanced design features of the small KSNP are as follows:

- Sixty year design life
- Integrated head assembly

– Permanent refueling pool seal assembly
– Digital plant protection and engineered safety features actuation systems
– Application of the leak-before-break concept to the piping with the diameter greater than 30 cm
– High pressure safety injection using charging system
– Common use of reactor makeup water tank and holdup tank for two units
– Single steam line on each steam generator
– Improved feedwater control logic
– Improved ex-core neutron flux monitoring system.

The small KSNP design complies with the licensing requirements, codes and standards which are based on a vast amount of design, manufacturing, construction, and operating experience. The applicability of specific codes and standards will be confirmed prior to construction of the plant.

The major plant parameters of the small KSNP are presented in Table 1 together with those of the 1000 MW(e) KSNP. The isometric view of the reactor coolant system (RCS) of the small KSNP is shown in Figure 1.

TABLE I. MAJOR PLANT PARAMETERS FOR SMALL KSNP AND 1000 MWE KSNP

Plant Parameters	Small KSNP	1000 MW(e) KSNP
General		
Electric Power Output, MW(e)	600	1050
NSSS Thermal Output, MWt	1714	2825
Reactor Core		
Number of Fuel Assemblies	121	177
Number of Fuel Rods per Assembly	236	236
Number of Control Element Assembly	45	73
Average Linear Heat Rate, kW/m	15.7	17.3
Reactor Coolant System		
Reactor Operating Pressure, MPa (abs.)	15.51	15.51
Reactor Vessel Inlet Temperature, ^{o}C	295.8	295.8
Reactor Vessel Outlet Temperature, ^{o}C	327.3	327.3
RCS Volume including Pressurizer, m^3	226.3	337.0
Pressurizer Volume, m^3	34.0	51.0
Reactor Vessel		
Total Height, m	13.05	14.64
Shell Outer Diameter, m	3.94	4.55
Steam Generator		
Secondary Pressure, MPa (abs.)	7.37	7.37
Feedwater Temperature, ^{o}C	232	232
Reactor Coolant Pump		
Rated Flow Rate, m^3/min	206.7	323.3
Rated Pump Head, m	93.0	102.7

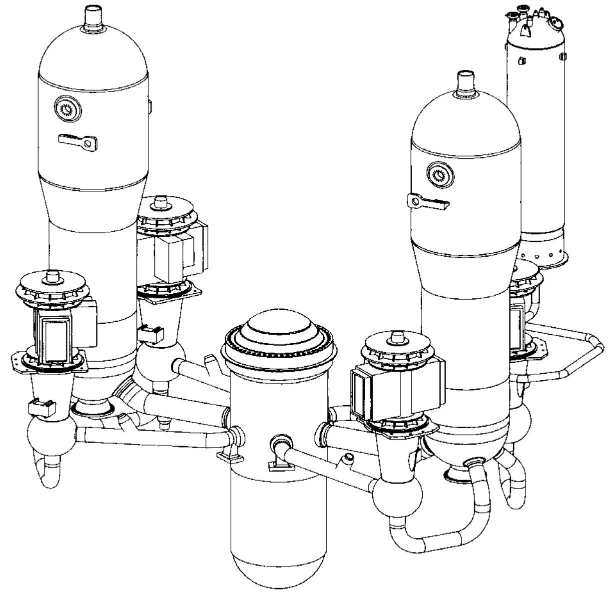

FIG. 1. Isometric View of Small KSNP Reactor Coolant System

3. SYSTEM DESCRIPTION

3.1. Reactor Coolant System

The reactor coolant system (RCS) arrangement is basically identical to that of the 1000 MW(e) KSNP having two closed loops forming a barrier to the release of radioactive materials from the reactor core to the secondary system and containment atmosphere. The major components are a reactor vessel, two steam generators, four reactor coolant pumps and one pressurizer connected to one of the hot legs and associated piping. The pressure retaining components of the reactor coolant pressure boundary are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The schematic diagram of the RCS is depicted in Figure 2.

The reactor core is composed of 121 fuel assemblies arranged to approximate a right circular cylinder. Forty five (45) control element assemblies including 8 spares are located to provide adequate shutdown margin and load maneuvering capability. Thirty one (31) in-core instrumentation assemblies are distributed over the core to provide the capability of core power tilt measurement, fuel misloading detection, ex-core detector calibration, etc.

The reactor vessel includes a vessel assembly, a removable closure head assembly, four inlet nozzles, and two outlet nozzles. The reactor vessel internals consist of the core support barrel assembly and the upper guide structure assembly. The sizes in radial direction of the reactor vessel and internals are determined to accommodate the reduced number of fuel assemblies. The overall axial lengths are not significantly reduced.

The steam generator is a vertical U-tube heat exchanger with an integral economizer. It is located at a higher elevation than the reactor vessel so that the elevation difference can create natural circulation to remove core decay heat following coastdown of all reactor coolant pumps. The number of steam generator tubes is reduced approximately proportionally to the power reduction, which leads to the reduced radial sizes. Each steam generator has one steam nozzle in comparison with two for the 1000 MW(e) KSNP.

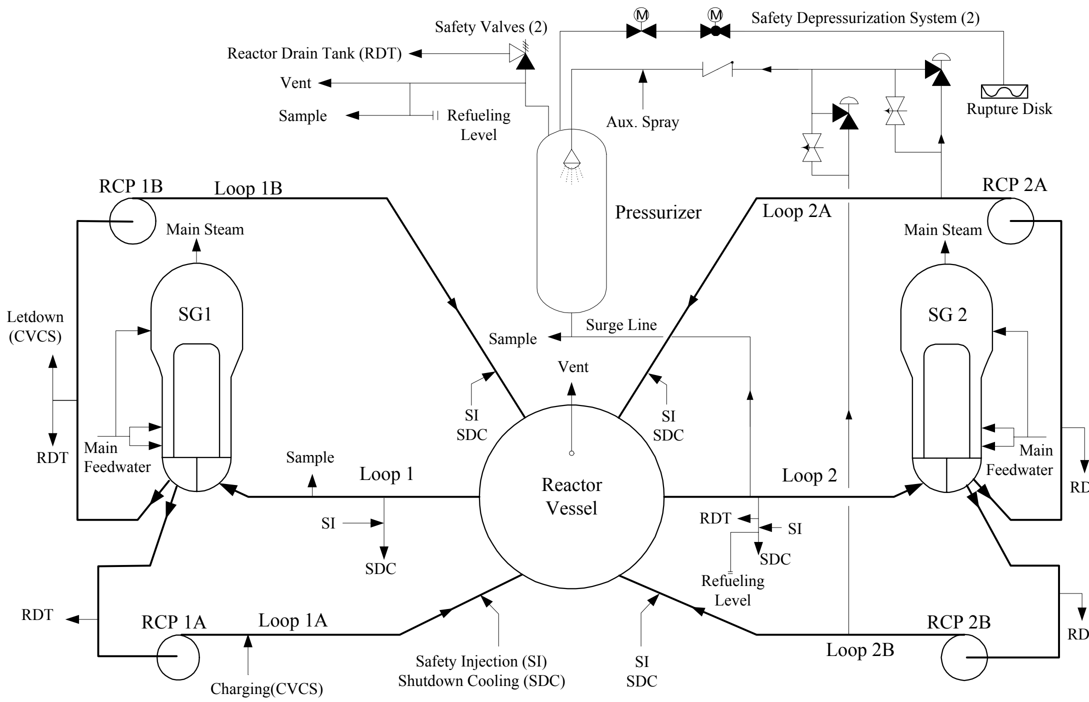

FIG. 2. RCS Configuration and Interconnection Systems of Small KSNP

The reactor coolant pump is a vertical, single stage bottom suction, horizontal discharge, motor-driven centrifugal pump. Each motor is provided with an anti-reverse rotation device. The rated flow rate is reduced according to the reduced power, which will downsize the impeller, diffuser, casing, motor, etc.

The pressurizer, connected to a hot leg via a surge line, controls the RCS pressure by maintaining the temperature of the pressurizer liquid at the saturation temperature corresponding to the desired system pressure by using heaters and spray. Using the volume sizing methodology, the total pressurizer volume is reduced approximately in proportion to the power ratio. For the overpressure protection of the RCS, two spring-loaded safety valves are provided whereas three in the 1000 MW(e) KSNP. The capacity of two safety valves is evaluated to be enough for the intended safety function. The pressurizer is also connected with two separate trains of the safety depressurization system.

3.2. Auxiliary Systems

The auxiliary systems include the safety injection system, shutdown cooling system, chemical and volume control system, and safety depressurization system.

The safety injection system provides emergency core cooling for an extended period of time in the event of a loss of coolant accident. It also provides inventory and reactivity control during other events such as steam line breaks and steam generator tube ruptures. The safety injection system consists of two redundant full capacity injection trains, four safety injection tanks, and associated valves, piping and instrumentation. For simplification, this system is designed to share the charging pumps of the chemical and volume control system for the high pressure safety injection function.

The shutdown cooling system is used in conjunction with the main steam system and the main or auxiliary feedwater system to reduce the temperature of the RCS in post shutdown periods from the hot shutdown operating temperature to the refueling temperature. It consists of two redundant full capacity trains, and two shutdown cooling pumps, associated valves, piping, and instrumentation. The containment spray pumps can be utilized to provide flow when the shutdown cooling pumps are not available.

The chemical and volume control system maintains sufficient reactor coolant inventory for core cooling, and provides sufficient flow to the reactor coolant pump seals for the RCS integrity. It also provides reactor water cleanup and process sampling for maintenance of the proper concentration of corrosion inhibiting chemicals in the reactor coolant. For design simplification, the reactor makeup water tank and holdup tank, both not directly related with the plant safety, are designed for common use by two units.

The safety depressurization system, one of unique design features of the KSNP, is adopted to rapidly depressurize the RCS to initiate feed-and-bleed operation of the RCS in conjunction with the safety injection system when both main and auxiliary feedwater systems are unavailable. It has two separate trains connected to the top head of the pressurizer and each train discharges to containment atmosphere through the rupture disc opening.

3.3. Instrumentation and Control Systems

The design features of the instrumentation and control (I&C) systems of the small KSNP are built upon the proven accomplishments of the 1000 MW(e) KSNP. It is based on existing proven analogue and digital technology and hardware, which ensures design flexibility and improves plant safety, availability, reliability and maintainability. The design makes extensive use of digital computers, programmable logic controllers, graphical display devices, and fiber-optic data communications. The digital plant protection system is specifically designed to meet the rigorous regulatory requirements for nuclear power plant protection systems.

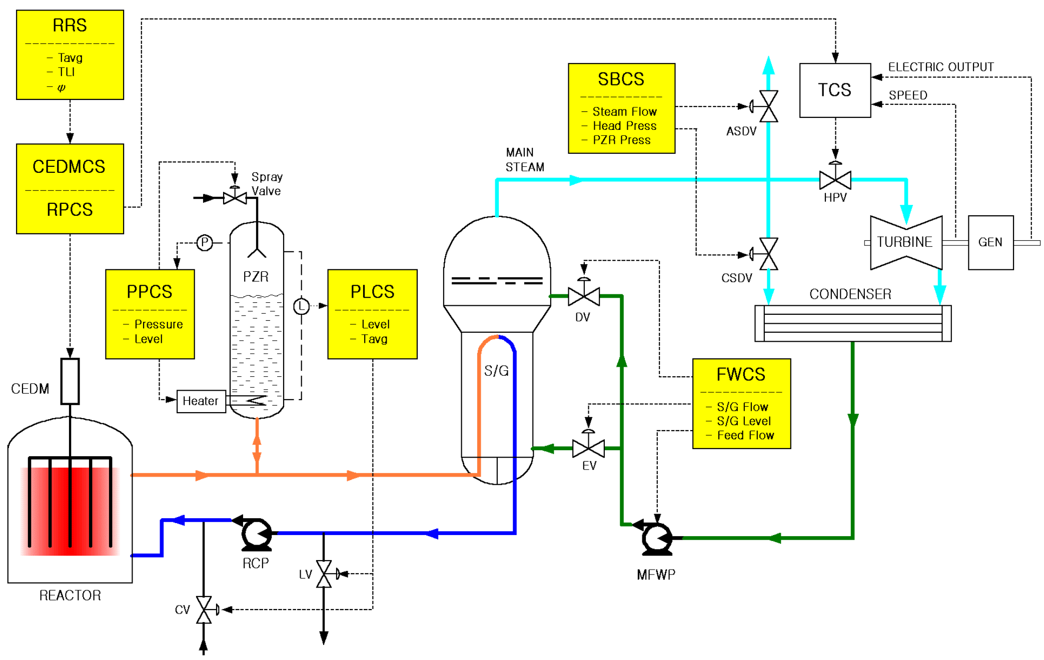

FIG. 3. Overview of the Plant Control System

The I&C systems consist of the reactor monitoring system, the plant protection system, the plant monitoring system and the plant control system. The reactor monitoring system monitors the reactor operating status by providing core cooling status, limiting conditions for operation, power distribution and neutron flux level information. The plant protection system is a digital system that monitors the selected safety-related plant parameters and initiates reactor trip and/or actuates engineering safety features upon detection of non-permissible plant conditions. The plant monitoring system improves the operator's situational awareness in both normal and abnormal modes of plant operation by providing advanced information processing, display and control with man-machine interface features. The plant control system enables the reactor to follow turbine load changes under normal plant operating conditions by using the reactor regulating system (RRS), control element drive mechanism control system (CEDMCS), pressurizer pressure control system (PPCS) and pressurizer level control system (PLCS), feedwater control system (FWCS), steam bypass control system (SBCS), and reactor power cutback system (RPCS), as shown in Figure 3.

4. SAFETY EVALUATION

The preliminary design transient and accident analyses are performed to verify that the plant parameters and component sizes are properly determined and to confirm that the safety systems are designed with sufficient margin. The analyses credit only safety grade systems in preventing and/or mitigating design basis events (DBEs) and consider a loss of offsite power after turbine trip concurrent with reactor trip and the worst single failure of a system or component. Also considered in the analyses are the effects of relative size reduction from the 1000MWe KSNP as well as the design changes adopted for the small KSNP such as a single steam line on each steam generator and the use of charging system for high pressure safety injection. Among the DBEs, the large break loss of coolant accident (LOCA) and the main steam line break (MSLB) accident are selected as the limiting events.

Figure 4 indicates that the analysis results for a large break LOCA satisfy the peak cladding temperature (PCT) limit (2200 ^{o}F (1204 ^{o}C)) specified by the regulatory acceptance criteria. The PCT for the small KSNP is about 170 ^{o}F (95 ^{o}C) lower than that for the 1000 MW(e) KSNP rendering more than 200 ^{o}F (110 ^{o}C) PCT margin. The increased PCT margin of the small KSNP is mainly due to a lower peak linear heat generation rate as compared to the 1000 MW(e) KSNP. The peak linear heat generation rates for the small KSNP and the 1000 MW(e) KSNP are 40.8 kW/m and 45.9 kW/m, respectively.

The major design parameters affecting the MSLB consequences are the flow restrictor area of steam nozzle, non-isolatable steam flow, safety injection flow, and control rod worth. The shutdown worth of 8.5% $\Delta\rho$ at all-rod-out condition is assumed in the evaluation. The maximum non-isolatable steam flow is to be assumed 9% of the maximum steam flow rate. Under these assumptions, a sensitivity study was performed with respect to the post-trip fuel performance degradation. As shown in Figure 5, a post-trip return to power occurs for the small KSNP while it does not for the 1000 MW(e) KSNP. The main reason for this difference is the increase in the steam flow restrictor area for the small KSNP caused by the adoption of single steam line on each steam generator. However, there are no fuel or cladding failures due to the post-trip return to power, thus core coolable geometry is maintained.

For the licensing of the small KSNP, the licensing experience of the 1000 MW(e) KSNP will be fully referred to, which itself has been downsized from 1300 MW(e) System 80 [3] and shown excellent performance for many years of commercial operation of four units in the

Republic of Korea. Where significant differences exist in size, relevant design verification test programs can be implemented to demonstrate the licensability of the downsized design. Model tests and operational tests in the field mainly constitute the programs. The former can be conducted in the detailed design phase, and the latter can be done during the plant startup test period after the completion of construction. Typical tests for the demonstration are the reactor flow model test, steam generator tube vibration test, reactor internals vibration test and reactor coolant flow rate measurement test.

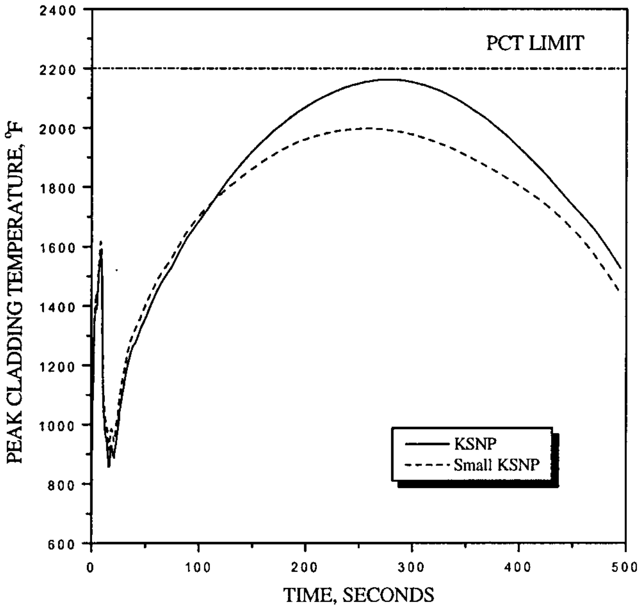

FIG. 4. Safety Margin during a Loss of Coolant Accident

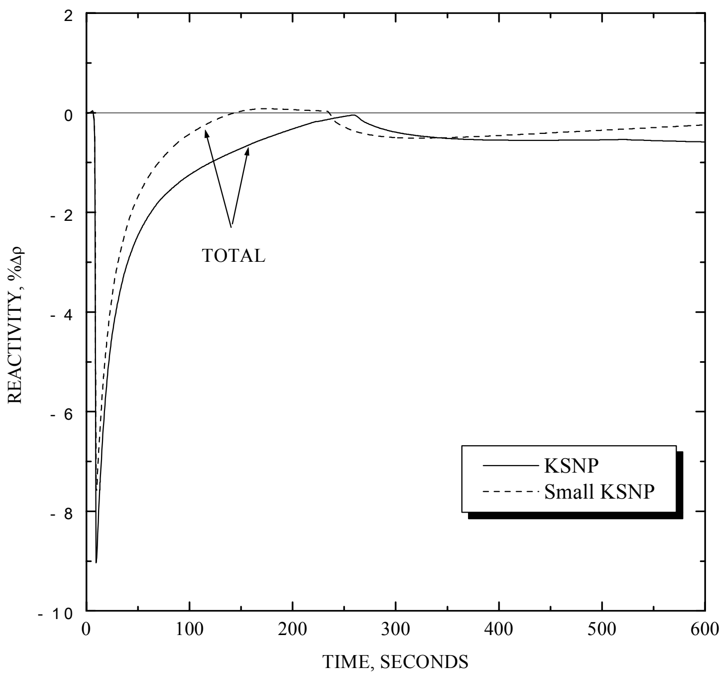

FIG. 5. Reactivity Transients during a Main Steam Line Break Accident

SUMMARY

The general design characteristics of the small KSNP being developed by KOPEC have been presented and compared with those of the KSNP currently in operation. Major components are downsized considering the power reduction, while the proven basic design concepts are maintained. Up-to-date design improvements and accumulated operating experience are incorporated into the system design. The preliminary accident analyses showed that safety systems are properly designed with sufficient margin and the plant safety are assured. Once the design is completed and the licensability is proven, the small KSNP is expected to be one of the most evolved and competitive 600 MW(e) class PWRs.

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BYPASS LINES TO REDUCE REACTOR COOLING-DOWN RATE DURING NATURAL CIRCULATION EMERGENCY CORE COOLING

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Abstract

This paper describes the main characteristics of a Natural Circulation Loop (NCL) and presents the first results of temperature and flow distribution when bypass lines that connect the cold source outlet with the hot source outlet are opened. The NCL was designed to study the hot-leg temperature effects caused by the opening of bypass lines connecting different hot-leg vertical positions. NCL is a thermal-hydraulic loop with a vertical electric heater and an immersion heat exchanger similar to those of advanced nuclear reactors. The heat exchanger outlet flow is measured with a magnetic flow meter. The bypass line flow is estimated through heat balance calculations. The first result, presented in this paper, shows that the flow direction in the bypass line is from the cold leg to the hot leg when the lower bypass line is opened and is in the opposite direction when the higher bypass line is opened. The paper also shows that the bypass line flow in the cold- to hot-leg direction is almost 10% of the total flow, which is enough to produce a substantial rise in the heater outlet temperature. This process can be used to reduce the cooling-down rate during natural circulation emergency core cooling, permitting a faster reactor restart.

Keywords: natural circulation, advanced reactors, thermal-hydraulics.

1. INTRODUCTION

Advanced nuclear power plant concepts cover different types of designs. The Advanced PWR is an interesting concept that uses natural circulation to remove the core residual heat. The emergency core cooling system is the system that achieves this residual heat removal. It is well known how necessary it is to build and operate a prototype or demonstration plant to bring a concept with much innovation to commercial maturity. It is also evident that there is the opportunity of introducing innovative features that require few development efforts but can produce substantial positive effects. These types of improvement need research, engineering and confirmatory testing before their use in a given plant.

Passive emergency core cooling systems of advanced PWR produces variable cooling-down rates depending on the residual heat rate. Occasionally, in the beginning of the cooling down process, when the residual heat decays substantially, reducing this cooling rate to restart the reactor in less time would be interesting.

In 1998 IPEN started a research project to study the effects of bypass lines in the hot-leg temperature of a system when operating in natural circulation. A Natural Circulation Loop (NCL) [1,2,3,4] was conceived and built for this purpose. In this thermal hydraulic loop the bypass lines connect the cold source outlet with the hot source outlet. Depending on the bypass to hot-leg vertical position connection, the bypass line flow can be either from the cold leg to the hot-leg or from the hot-leg to the cold leg. If the Main Bypass line flow follows the cold-to-hot direction, the net flow through the heater will be smaller. The heat exchanger flow will be the sum of the heater flow with the bypass line flow. The heater outlet temperature will be higher compared with the temperature without bypass flow.

This paper consists of four sections. The first section is this introduction. The second section describes the main characteristics of NCL. The third section describes the experiments and shows the first results of the temperature and flow distribution when the bypass lines that connect the cold source outlet with the hot source outlet are opened. The fourth section presents the conclusions and discusses future works.

2. NATURAL CIRCULATION LOOP DESCRIPTION

Fig. 1 presents a schematic layout of the Natural Circulation Loop (NCL) which resembles an Advanced Pressurized Water Reactor Decay Heat Removal System. The NCL has an electric heater that it is the hot source of the system, and a heat exchanger, that is the heat sink. Cold water coming from an elevated water reservoir is supplied to the water tank by gravity. The NCL has a water circulating pump installed to allow specific operations for calibration and measurement of the hydraulic characteristics of the system. A magnetic type flow meter is installed in the main circuit line. The magnetic flow meter was chosen to measure water flow rates below 0.1 kg/s, without flow interference. A globe valve controls the secondary cooling water flow with the aid of a variable area flow meter. As NCL was designed to study the hot-leg temperature effects caused by the opening of bypass lines connected in different hot-leg vertical positions, it was provided with a main bypass line with a motor operated valve (bypass valve). This main bypass line is connected with the hot-leg in two different positions: a High bypass connection and a Low bypass connection. The main NCL circuit pipe consists of copper with 22 mm O. D. and 0.6 mm thickness. All connections are welded and the main components are fitted with 3/4 in threaded connections.

To avoid unnecessary loss of head, the valves in the main loop lines are ball type. Fig. 1 shows the 24 thermocouples distributed in the experimental loop. There are nineteen 0.5 mm K Type thermocouples and five 1.5 mm T Type thermocouples.

The electric heater has three "U" type electrically heated elements of stainless steel clad and a non heated central rod, making a hexagonal array with a heat transfer area of 0.474255 m^2. The heating elements are of 14.8 mm in diameter and 1.7 m in length. The shell diameter is 2 ½ inches. The heater design power is 10 kW but its maximum operating power was limited to 3.3 kW in the natural circulation tests. A specially designed rectifier receives a 0-10 DCV control signal from any kind of external font and controls the electric power. The power control signal, flow meter signal and thermocouples' signals are connected to a AT-MIO-16E National Instruments Data Acquisition Board [5] installed in an Intel Pentium type P.C. (Fig. 2). The 26 signals are multiplexed and registered in intervals of 16 seconds of the data acquisition period.

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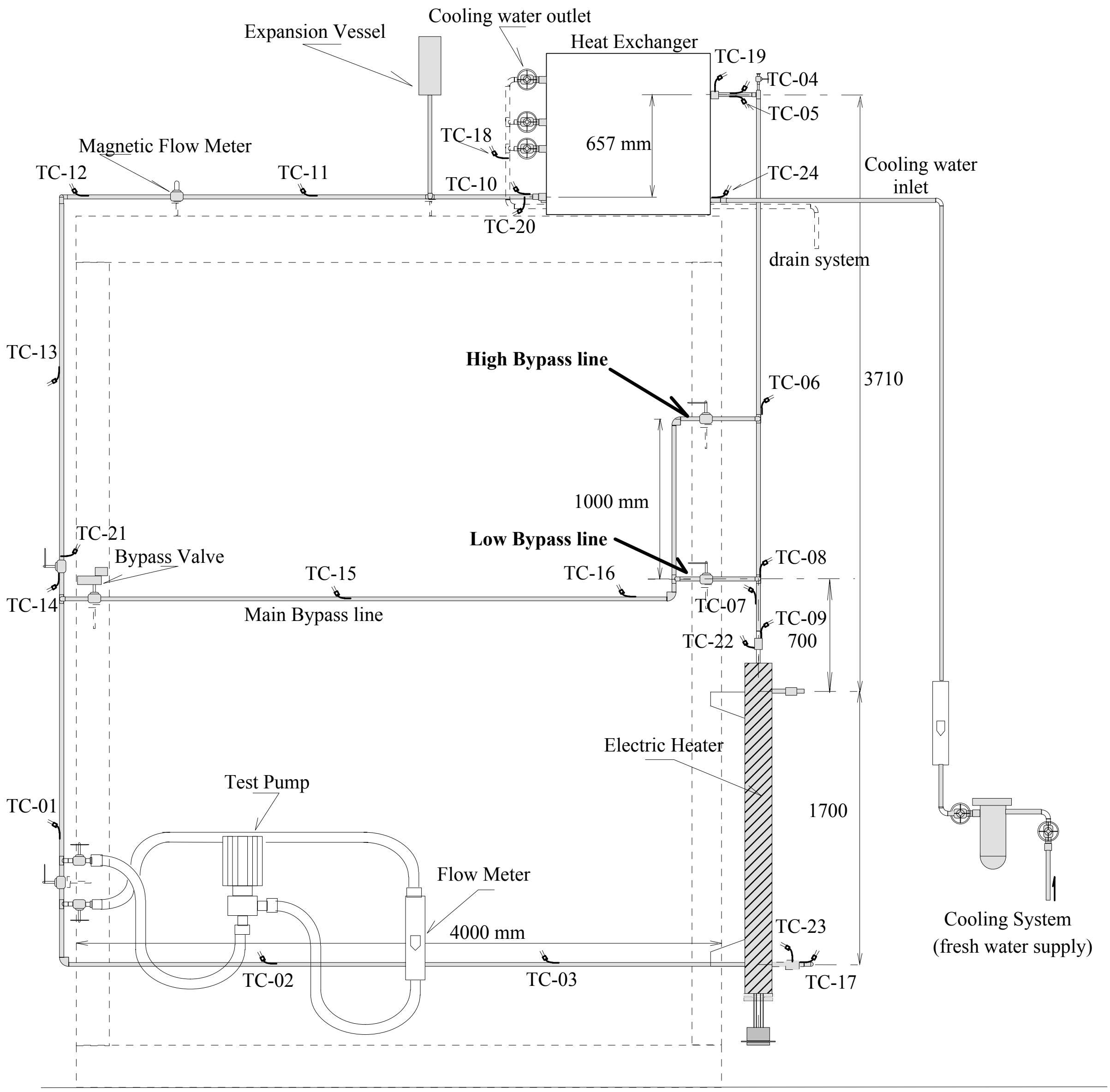

FIG.1. Natural Circulation Loop layout.

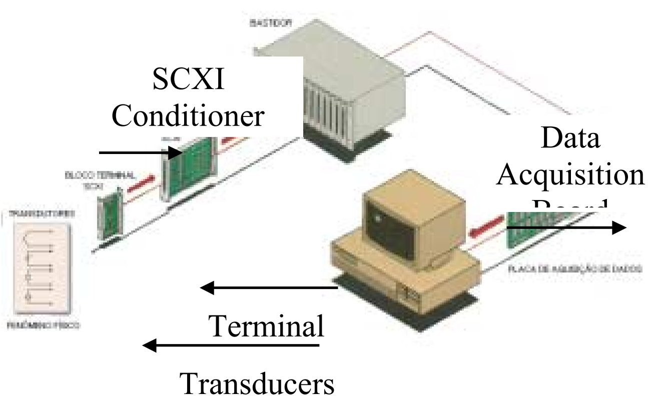

FIG. 2. Data Acquisition System.

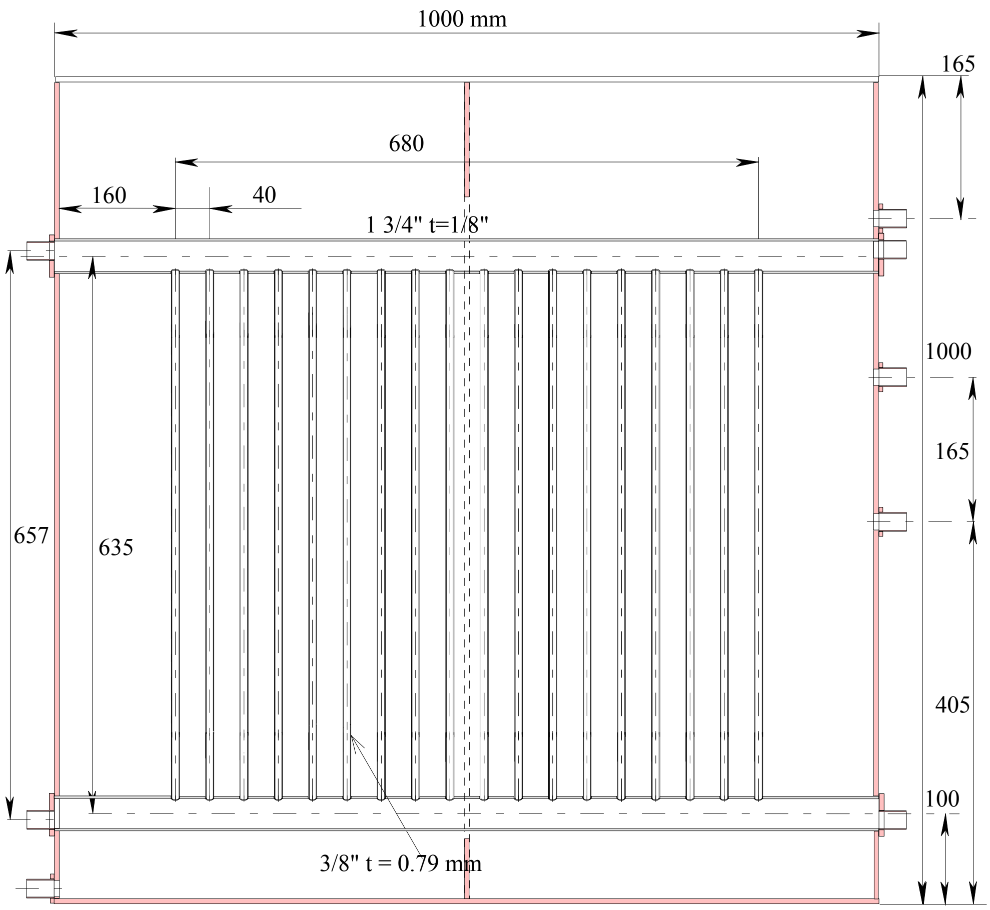

FIG. 3. Heat Exchanger Cross Section.

The heat exchanger (Fig. 3) is a reservoir of 0.202 m^3, rectangular, and made of copper. It has two 1.75 inches in diameter headers and eighteen 3/8 inch vertical tubes. The total heat transfer area, based on the external diameters is 0.621 m^2. The heat exchanger was designed based on naval and on AP-600 designs [6].

3. NATURAL CIRCULATION EXPERIMENTS

This section describes the first set of experiments with the bypass lines opening. In the first one of these experiments, the low bypass line was opened twenty-two minutes after the start of power operation. The second experiment followed the first one by the closure of the low bypass line and with the high bypass line opening. The results of the first experiment are compared with another experiment with the external loop steady state operation without bypass flow. To allow this comparison the results were superposed in the time scale. The second experiment initial time was set immediately after the end of the first one.

The data acquisition during the first experiment was initiated a few minutes before the heater was on in a partial power of 1250 W. After twenty-two minutes the bypass line was opened. Ten minutes later the power was increased to ≈ 2300 W. The electric power and mass flow rates registered are shown in Fig. 4. The heat exchanger mass flow rate was measured with the magnetic flow meter. The heater and bypass line mass flow rates were estimated through mass and heat balance as explained through Fig.5 and equations below:

$$m_{Ht}\,h_{Ht} + m_{Bp}\,h_{Bp} = m_{HE}\,h_{HE} \tag{1}$$

$$m_{Ht} + m_{Bp} = m_{HE} \tag{2}$$

where m_{Ht} is the heater mass flow rate, m_{Bp} the bypass line flow rate, m_{HE} the heat exchanger mass flow rate, h_{Ht} is the heater outlet enthalpy, h_{Bp} is the bypass line enthalpy and, h_{HE} is the heat-exchanger inlet enthalpy.

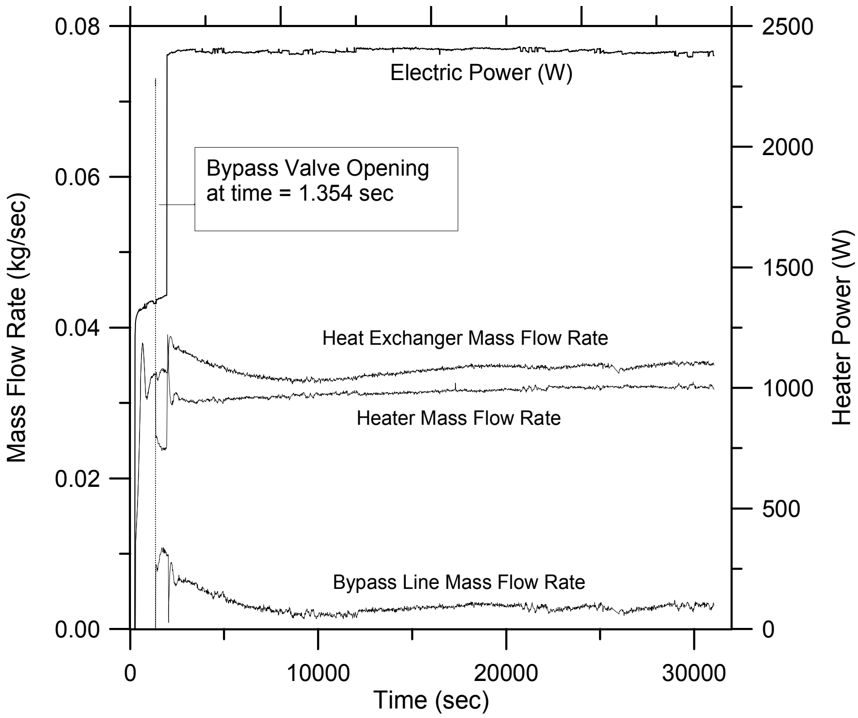

FIG. 4. Heater Power and Mass Flow Rates - First Experiment.

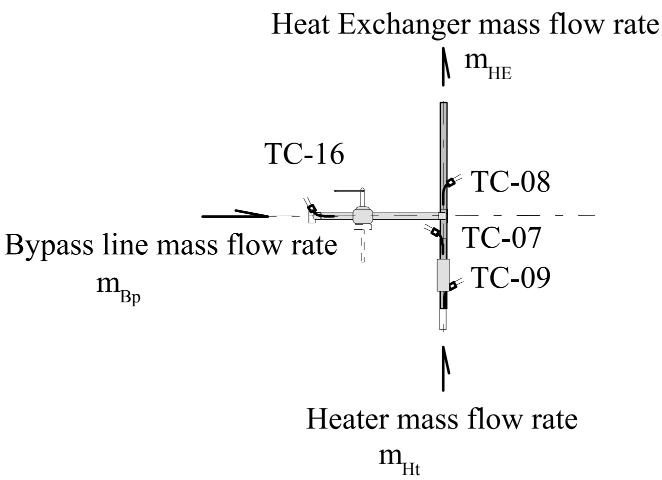

Figure 5 - Node for Heat and Mass Balance.

Due to the natural circulation processes characteristic of the NCL loop, the temperature and mass flow rates changes are very slow, then the errors associated with heat and mass balances due to the transient conditions are small. Because of this one can assume that the bypass mass flow rate estimate will be close to the real one. Fig. 6 shows the bypass mass flow rate of Fig. 4 as a percentage of the total flow, the heat exchanger mass flow rate. It can be observed that, as the initial bypass line temperature is small, the initial bypass flow grows quickly to 30% of total flow, then falls to approximately 10%. This low fraction of mass flow is enough to produce a 15-degree C temperature rise in the heater outlet as can be seen in Fig. 7, where the temperature result of this transient is superposed to the results of an equivalent transient without the bypass line opening. The power-step amplitude and timing in the two experiments were not the same but the steady state power was the same. In the experiment one, the bypass line was opened 1354 seconds after the start of data recording causing a perturbation in the outlet heater temperature.

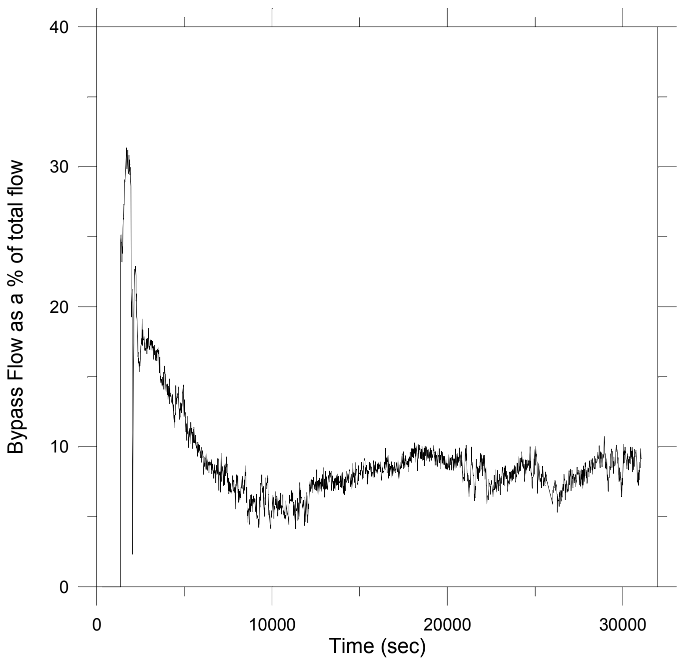

FIG. 6. Bypass Flow Rate as a Percentage of Heat Exchanger Mass Flow.

The second experiment was in sequence of the first one and lasted 14:46 hours. Fig. 8 shows the electric power and mass flow rates registered during this time. In this experiment the high bypass line was opened at time = 250 seconds. Because this experiment followed the first one, the temperature rapidly reached the steady state condition, but the experimental setup was kept in operation to observe the NCL power and water supply stability, this can be observed in the heat exchanger temperatures of Fig. 9. Fig. 8 shows that the total flow rate is through the heater instead of the heat exchanger. Now the bypass line flow rate is stipulated to be negative. Fig. 9 shows that the experiment was very stable. The first hour of this experiment bounds the greatest changes in flow and temperatures. Fig. 10 shows the temperature behavior around the bypass line and Fig. 11 shows the bypass percentage of mass flow rate, which reaches a mean value of 18% in the first hour. The most important observation in this experiment was that, as the heater mass flow rate was almost not affected by the bypass opening, the heater outlet temperature did not change. The heat exchanger outlet temperature had changed but the heater inlet temperature was compensated with the bypass water mixture. Here the main component affected was the heat exchanger.

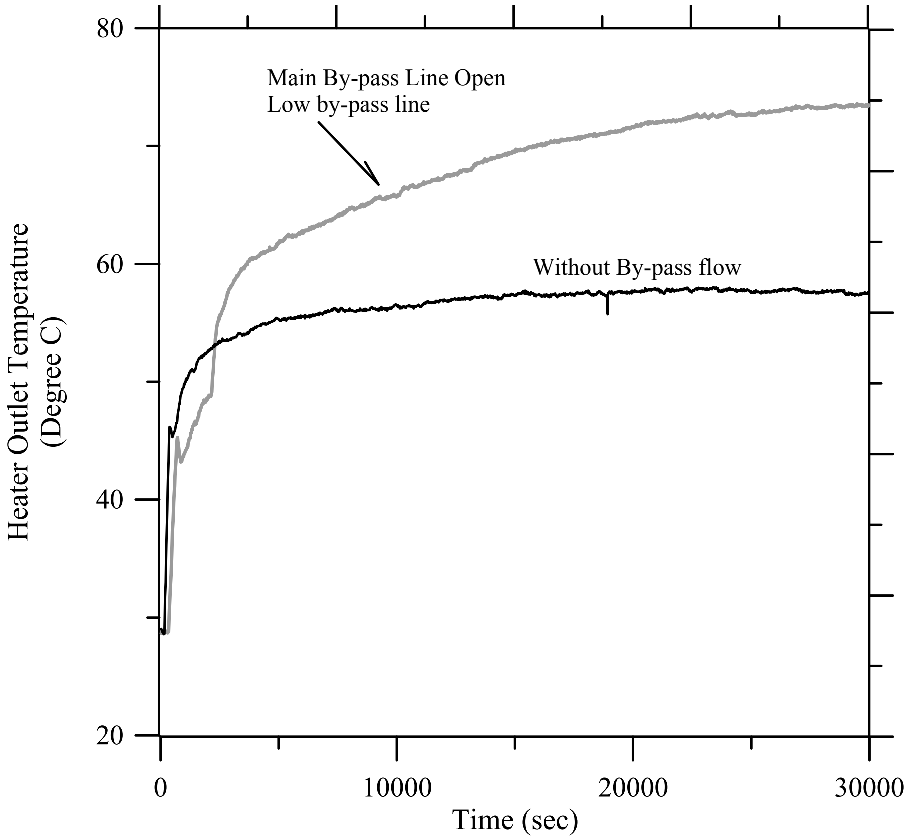

FIG. 7. Heater Outlet Temperature Results.

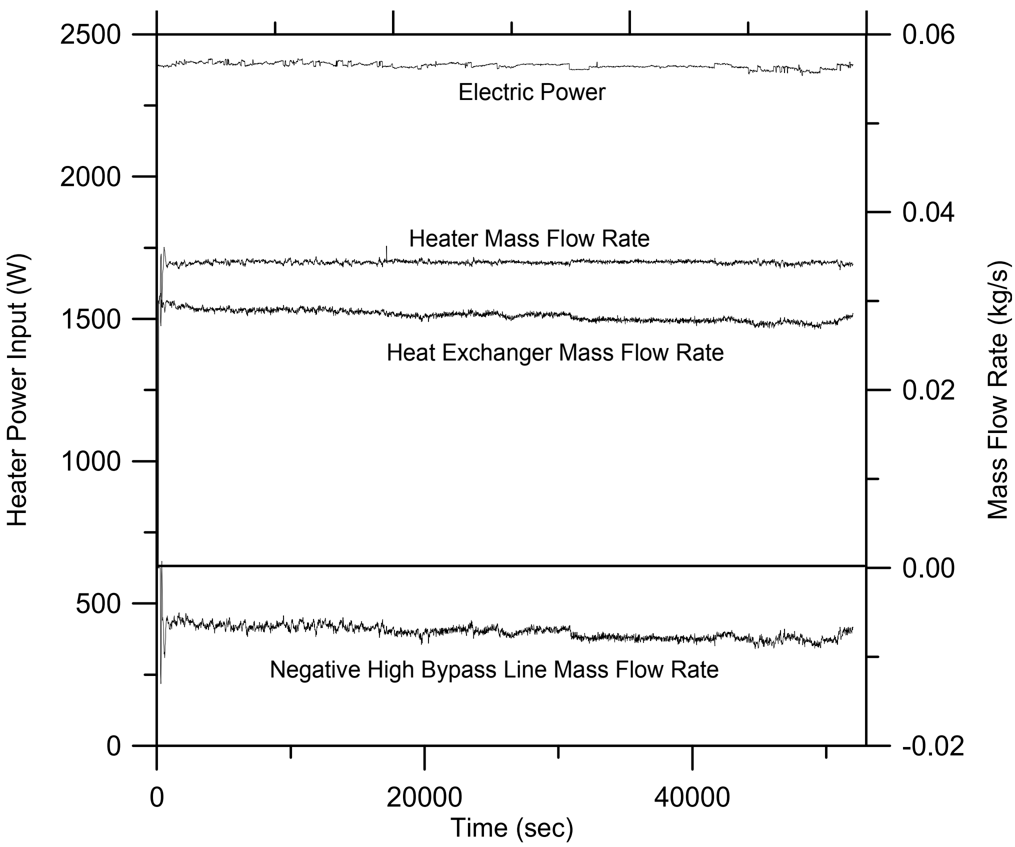

FIG. 8. Heater Power and Mass Flow Rates - Second Experiment.

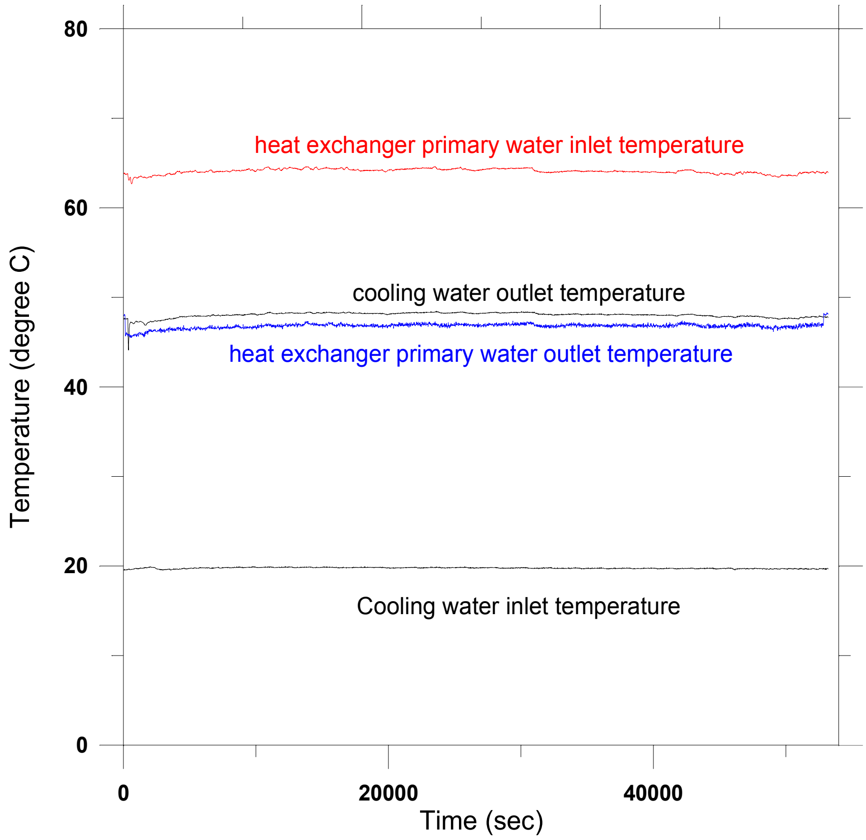

FIG. 9. Main components temperature behavior.

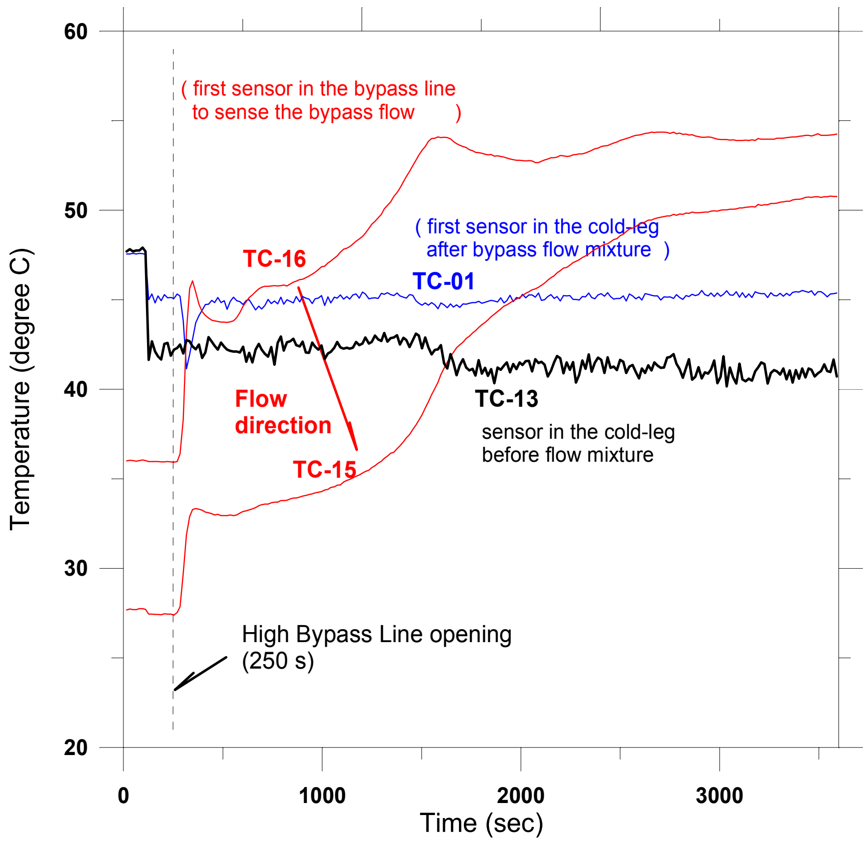

FIG. 10. Bypass line temperatures behavior.

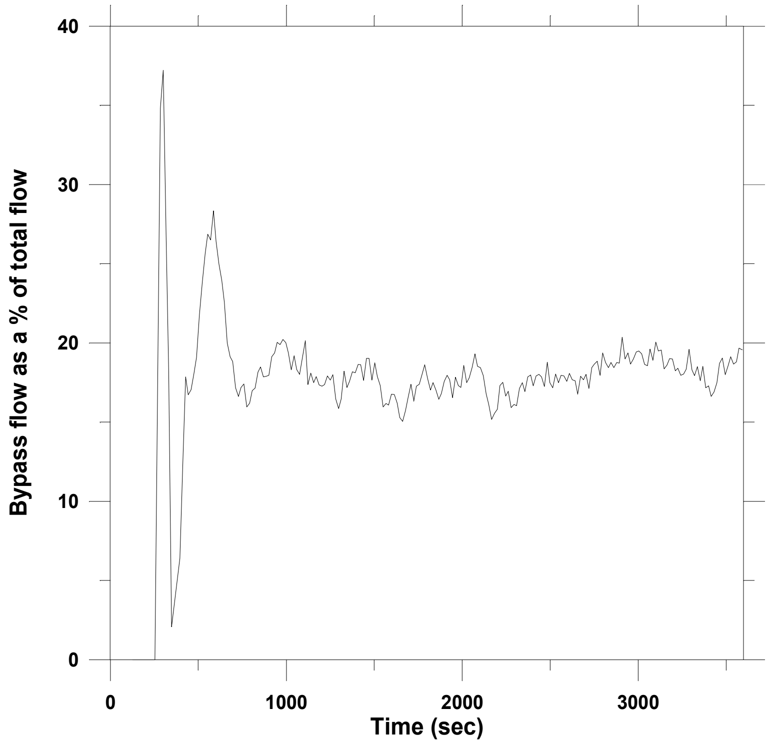

FIG. 11. Bypass line percentage of flow.

CONCLUSIONS

The first sets of experiments with the opening of bypass lines in the Natural Circulation Loop (NCL) were analyzed and show that bypass lines can be used to control the outlet temperature of the hot source in a natural circulation system.

The results presented in this paper, shows that the flow direction in the bypass line is from the cold leg to the hot leg when the lower bypass line is opened and is in the opposite direction when the higher bypass line is opened.

It was observed that the bypass line flow in the cold- to hot-leg direction is almost 10% of the total flow, which is enough to produce a substantial rise in the heater outlet temperature. This process can be used to reduce the cooling-down rate of a reactor vessel during natural circulation emergency core cooling, permitting a faster reactor restart. The temperature changes will be higher in the emergency cooling system.

Depending on the bypass to hot-leg vertical position connection, the bypass line flow can be changed to the hot-leg to the cold leg direction. If the main bypass line flow follows the cold-to-hot direction, the net flow through the heater will be smaller; the heat exchanger flow will be the sum of the heater flow with the bypass line flow. The heater outlet temperature will be higher compared with the temperature without bypass flow. If the vertical connection position of the bypass line with the hot-leg is higher, the flow changes its direction and, even if this flow is bigger than that observed in the other case, the heater conditions will not suffer great changes. In the second experiment presented, a bypass flow equivalent to 18% of the total flow was observed.

In the future experiments initial and boundary conditions will be controlled to be the most similar as possible. The convenience of application of power steps will be carefully analyzed before starting any other experiment. It was not commented before but, an uncertainty of 16% in the flow measurements was observed, while the expectation was only 10% uncertainty.

A numerical model is going to be developed to aid the experiments planning. This model will be used in future works to find the vertical connection position of the bypass line with the hot-leg where the flow direction changes.

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AN INNOVATIVE CONCEPTUAL DESIGN OF THE SAFE AND SIMPLIFIED BOILING WATER REACTOR (SSBWR)

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Abstract

Small-sized water reactors SSBWRs have been developed for multi-purpose energy sources, using innovative concepts based on reliable BWR technology and the former medium-sized BWR design. Design targets of the SSBWRs are set as follows: (a) simplicity of operation and maintenance, (b) multi-purpose energy supply, (c) passive safety, (d) cost reduction. A representative concept of SSBWRs the following characteristics: a 20-year core life, an indirect cycle and a passive core safety system with infinite grace period. Operability and maintainability can be largely improved by a super-long life core using heavy water and a reduction in the number of active components such as pumps. An indirect cycle with high temperature secondary steam is used for a multi-purpose energy supply. Natural coolant recirculation and gravitationally induced safety systems improve the reliability of plants with complete passive safety. The construction cost can be reduced by compact design of the reactor pressure vessel (RPV), the primary containment vessel (PCV) and the simplified BOP system using the indirect cycle.

1. INTRODUCTION

Innovative reactors need to keep up with new requirements in the new energy era. Nuclear reactors have been developed with higher power generation capacity and reduced power generation cost per unit electric power. In recent years, however, the social and market circumstances of the atomic energy industry have changed and diversified.

A wide variety of reactor capacities has been expected for various market needs. A distributed energy source is needed for developing areas to overcome poor electric transfer networks and for developed areas to minimize investment risk under liberalization of electric power generation and sale. The utilization of atomic energy will also be varied and diversified. Multi-purpose energy sources of heat and electricity will be needed in both remote areas and cities, which is applicable to district heat sources for air conditioning, process heat sources such as seawater desalination, co-generation of heat and electricity and so on. Maintainability and operability are needed to maintain inherent safety without placing experts on distributed sites. Inherent safety is required at the very least to reduce operators' burdens with no additional operations and at the very most with no evacuation of the public surrounding the distributed sites, even in the case of severe accidents. A small-sized reactor is flexible and it is now being focused on as a suitable reactor for these situations. It is necessary to prevent nuclear proliferation for distributed reactors which will be constructed around the world. Improvements in economy and reductions of nuclear waste should be made to keep atomic energy in competition with other energy sources.

A small- and medium-sized reactor HSBWR (Hitachi Small BWR) with a rated capacity of 600MWe has been already conceptually designed [1]-[5]. The design bases were to improve seismic resistance for various location conditions, to introduce inherent safety, to improve maintainability and operability and to decrease capital costs. The concepts used standardization of the reactor building, natural circulation for core cooling and a passive safety system for decay heat removal with a 3-day grace period at the time of an accident. The operation cycle was 24 months and the components and systems in the reactor were simplified by eliminating pumped recirculation systems and pumped emergency core cooling systems. The capital costs of construction were reduced by eliminating active components and systems contained by a small PCV and reactor building.

Small-sized water reactors SSBWRs have been developed to meet various new needs. They are based on HSBWR and ABWR technology with addition of several innovative advantageous concepts applicable to various market needs. A super-long life core of 20 years and RPV renewal with no exchange of fuel assemblies were applied to prevent nuclear proliferation and to improve operability and maintainability remarkably. An indirect cycle with compact heat exchangers was used for a multi-purpose energy supply using a high temperature secondary coolant. A passive safety system with infinite grace period of heat removal was adopted in order to ensure no need for evacuation of the public in the event of a severe accident. In this paper, a representative concept of SSBWR with 20-year core life, indirect cycle and passive core safety system with infinite grace period is described herein.

2. REACTOR CONCEPTS AND SPECIFICATIONS

The design features of SSBWR are compared with those of the ABWR in Table 1. The reference output powers are 150MWe and 434MWth, suitable for a distributed small energy source. The design base points are as follows.

1. Simplicity of operation and maintenance
   - an operation life of 20 years
   - the reduction of active components
   - prevention of nuclear proliferation
   - reduction of waste
2. Multi-purpose energy supply
   - co-generation of heat and electricity with high efficiency
3. Passive safety
   - neutronics insensitive to thermal hydraulics change in an accident
   - naturally induced safety systems
   - infinite grace period of accident
   - no evacuation of the public, even for a severe accident
   - no large break LOCA (loss of coolant accident)
4. Cost reduction
   - compact design of RPV, PCV and BOP system

A 20-year operational life is selected to take into account maintainability and material life under irradiated conditions. Long life operation of the core can be achieved by using a less moderated neutron spectrum by selecting fuel lattice, fuel material and/or coolant. Heavy water is selected as the coolant by considering additional effects of the reduction of control rod drives (CRDs) and neutronics characteristics which are insensitive to accidents. A direct cycle as used in normal BWRs can be applied to a multi-purpose energy supply, but an indirect cycle is chosen because of flexibility in application to co-generation using high temperature secondary steam. An infinite grace period can be achieved using the heat pipe system with external heat sinks because the heat pipe can transfer decay heat in a reactor to outer spaces by using evaporation and condensation of steam and water while keeping the driving fluid volume constant under low pressure.

TABLE I. COMPARISON OF DESIGN FEATURES OF SSBWR AND ABWR

	SSBWR	ABWR
Reactor Type	Indirect Cycle BWR	Direct Cycle BWR
Thermal Power	434MWth	3926MWth
Electric Power	150 MWe	1356 MWe
Operation Life	20 years	1～2 years
Moderator	Heavy/Light Water	Light Water
Primary Cooling System	Natural Convection	Forced Convection
Pressure (Primary/2nd)	12MPa/7MPa	7MPa
Safety System	Passive	Active

3. CORE DESIGN

Super-long life operation of 20 years can be achieved with a less moderated neutron spectrum by using heavy water as a coolant and a triangular tight fuel lattice in core fuel assemblies of enriched UO_2 (Fig.1). A less moderated neutron spectrum improves the internal conversion ratio which reduces burnup reactivity and also introduces a small dependency of core reactivity on coolant void fraction which realizes mild transient behavior and small reactivity swing between hot and cold state as shown in Fig.1. There are two kinds of fuel assemblies. The first type of fuel assembly (Type I) has no guide thimbles for cluster control rod insertion and consists of 397 enriched UO_2 fuel rods. The second type of fuel assembly (Type II) has 30 guide thimbles and 367 fuel rods. There is no channel box in either type of fuel assemblies. Coolant flow is separated by thin plates being settled at the core support as a honeycomb. Fuel assemblies are loaded between these plates. There are 7 control rod drivers; 6 control rod arms are branched from one control rod drive and each arm has 30 cluster control rods which are inserted into Type II fuel assembly. Gadolinium is mixed in 6 fuel rods of each type of fuel assembly to maintain sub-criticality in case light water is injected into the core. The core consists of 60 Type I fuel assemblies and 91 Type II ones. The equivalent core diameter is about 3.4m and the core height is 1.2m. The discharge fuel burnup is about 65GWd/t. The core can decrease large excess reactivity due to its super long life and one-batch core operation and it can decrease the reactivity swing between normal operation and cold shutdown states by using heavy water and supplemental burnable poison. The excess reactivity is about 7.8% Δ k/kk'.

The increase of coolant void reactivity due to core burnup can be compensated by diluting heavy water with light water gradually. The example of a dilution scheme of water is shown in

Fig.2. Initial primary coolant is just heavy water; light water is gradually mixed with heavy water to become 70% of the coolant after 16 years. The small excess reactivity can reduce the necessary number of CRDs and simplified RPV internals and lower cost can be achieved.

4. REACTOR INTERNALS

Fig.3 shows the RPV and reactor internals. There is no liquid/vapor separation system above the core in the RPV and steam generators (S/Gs) are inserted into the steam region so that a compact indirect cycle can be applied using the high temperature secondary loop for a multi-purpose energy source. Secondary pressure is maintained at 7.1MPa to maintain thermal efficiency the same as that of the BWR, but primary pressure is increased to 12MPa to reduce S/Gs volume under the minimum critical power ratio (MCPR) condition. Natural recirculation is used for core cooling, which eliminates recirculation pumps. The RPV with 4.2m diameter and 14m height is made compact by eliminating pumps and using small S/Gs with high condensation heat transfer. There is no feed water line and no primary coolant steam lines so that the probability of a large LOCA is greatly lowered.

Thermal power can be controlled by flow rate of primary coolant using primary pumps for the ABWR, but there is no primary pump for the SSBWR. Thermal power, therefore, can be changed by the water level. The chimney has a lot of holes and the natural recirculation flow rate is changed linearly with the water level where opening rate is ～2%. The flow rate can be decreased by opening the holes, but the natural recirculation capability is sufficient to cool down the core with a chimney of 5m. MCPR is evaluated to be over 1.3 by the revised CISE formulation [6] for the triangular tight fuel lattice by using ISCOR and SILFEED codes [7]. Natural recirculation is maintained by using the holed chimney, even if the water level is lowered at LOCA.

The internal S/Gs are located above the core and steam generated at the core is condensed. Condensed water is returned to the core through an annular down-comer outside the chimney. The S/Gs are shell-tube type heat exchangers with the shell side of primary coolant and the tube side of secondary coolant. The secondary coolant flows out into the steam turbine and the outlet temperature of the secondary coolant is 560K at 7.1MPa, which is almost the same as the ABWR. The S/Gs are compact due to high heat transfer of condensation at the outer wall of the tube and evaporation at the inner wall of the tube. The S/Gs are also used for decay heat removal of core with natural convection of primary coolant.

5. SAFETY SYSTEM

Reliability can be enhanced by the passive safety system. Fig.4 shows the PCV and safety system of the SSBWR. The PCV contains a suppression pool above the RPV, which allows gravitational water injection to cool down the RPV. In an accident, the RPV is automatically depressurized and the water injection system is operated gravitationally to cool down the core. Even if the core is melted down, water injected gravitationally into the outside of the RPV would keep the core within the RPV. Discharged steam in the PCV is condensed in the suppression pool and the heat is transferred from the suppression pool to the outer air by using a heat pipe. The system can achieve the infinite grace period of heat removal while keeping the outside of the PCV isolated. In an accident, therefore, the safety system is completely passive and the reactor is kept without severe damage to satisfy the design point of no need for evacuation of people around the reactor site.

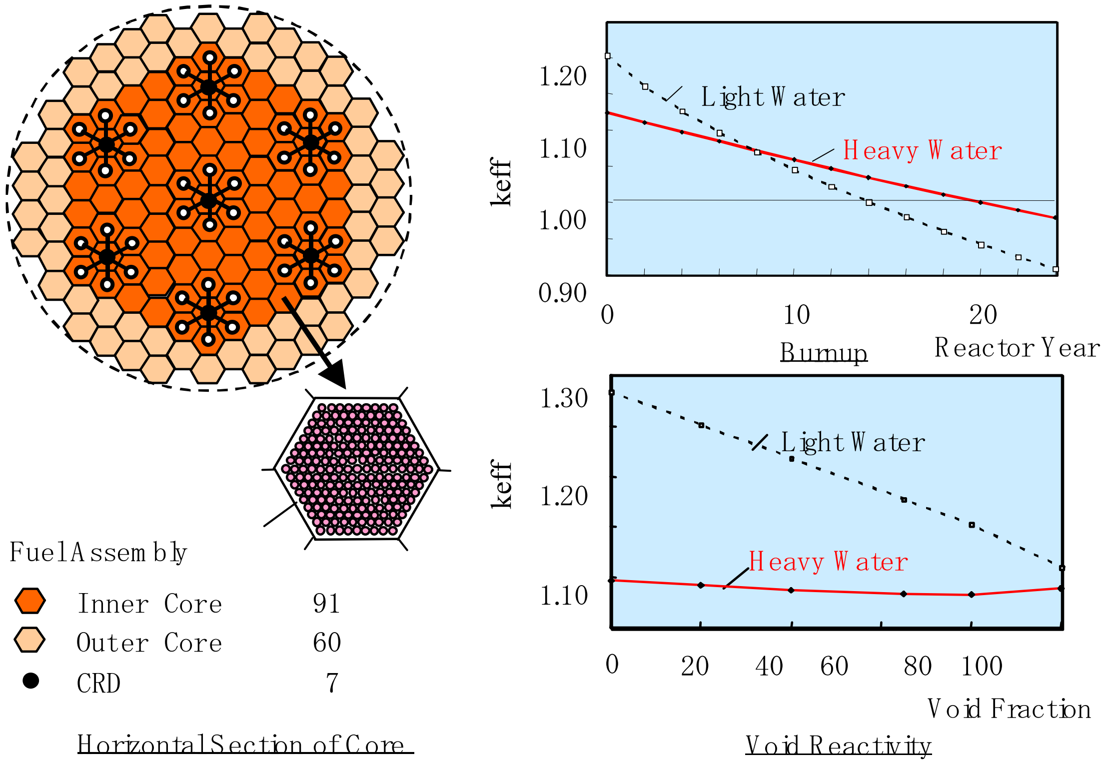


Fig.1 Super-Long Life Core

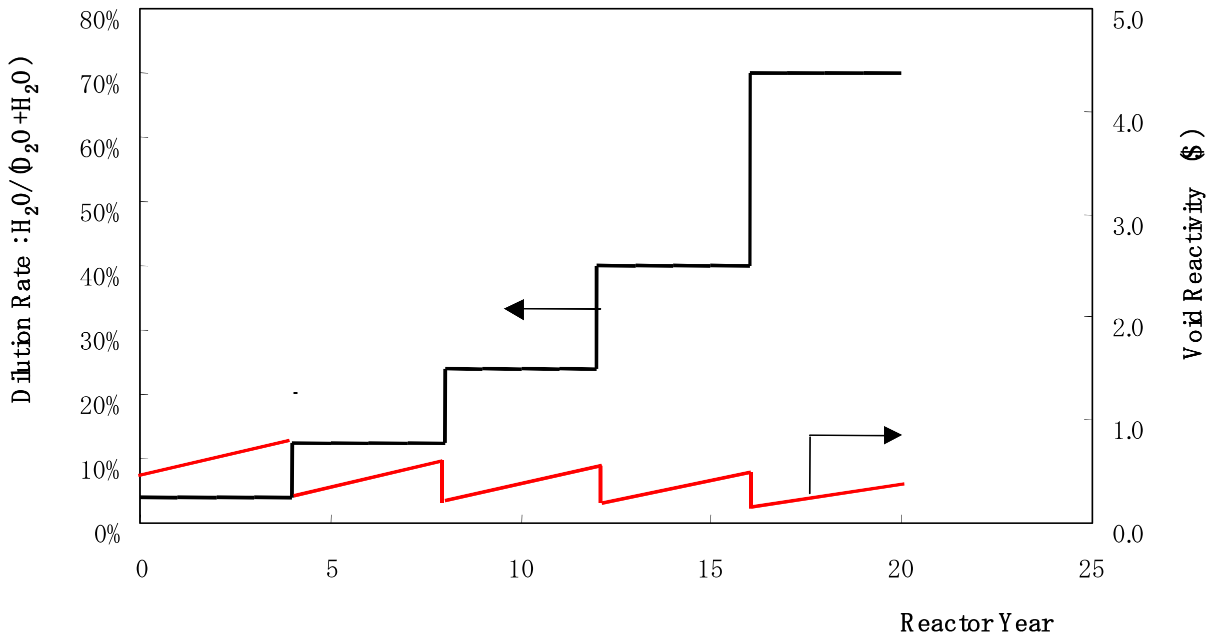


Fig.2 Example Dilution Scheme of Water to keep a Small Void Reactivity

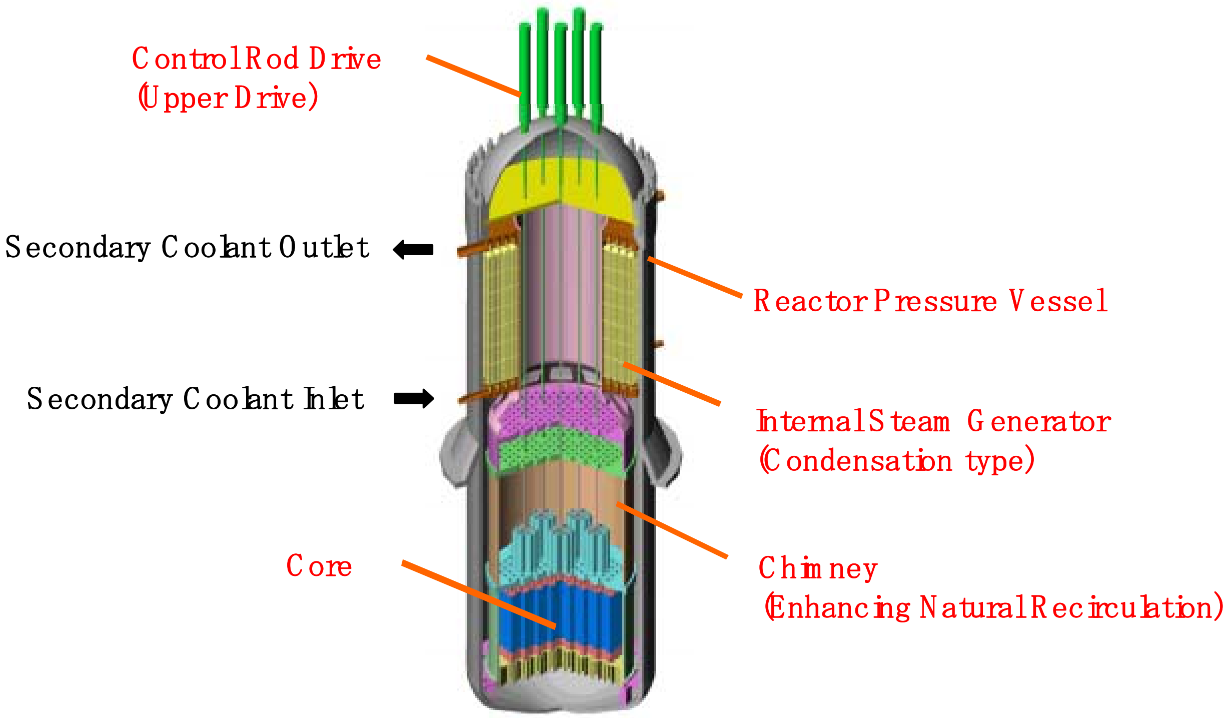


FIG. 3. Reactor Pressure Vessel and Internals

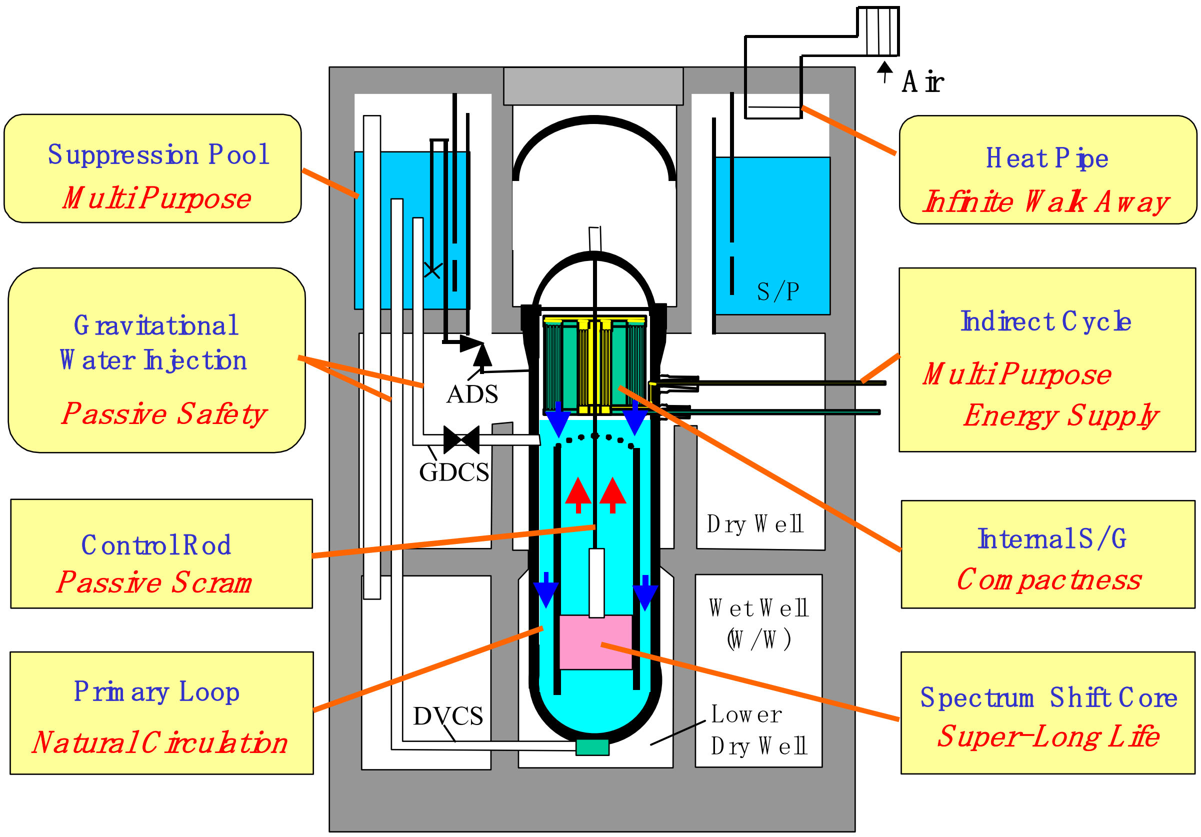


FIG. 4. Primary Containment Vessel and Features

5.1. Thermal Hydraulics in Small LOCA

The natural recirculation BWR has configurations for postulated LOCAs by pipe breaks, because there are no large diameter pipes below the top of the core. Moreover, for the SSBWR, there is no large feed water pipe or primary coolant steam lines which are connected to the RPV. The phenomena investigated for LOCAs are only small piping breaks such as the clean-up line of water (CUW). The emergency core cooling systems (ECCSs) are equipped with the automatic depressurized system (ADS) and the gravitational driving coolant injection system (GDCS), which are designed to achieve the conditions of no core uncovery and no fuel cladding heatup during LOCAs. After the system pressure decreases with the ADS operation, the GDCS is activated.

The PCV is divided into three stages, the upper suppression pool, the middle dry well and the lower wet well. The gas region above the upper suppression pool is connected to the lower wet well by vent pipes. Here, the wet well at the bottom of the PCV is a closed confined room where non-condensed gas is contained. In the small LOCA, pressure is increased in the PCV and non-condensed gas initially confined in the dry well flows into the suppression pool. Then discharged steam in the dry well enters the suppression pool where steam is condensed. Temperature of the suppression pool is increased and evaporation starts to occur. Non-condensed gas evacuates into the wet well through vent pipes and steam fills in the gas region above the suppression pool. Heat pipes which connect the gas region above the suppression pool and the outer space of the PCV can remove heat inside the PCV effectively due to high heat transfer of condensation in the gas region above the suppression pool. Decay heat gradually balances with cooling of heat pipes and S/Gs inside the RPV. Once the GDCS is activated, steam is generated at the core, passed through the dry well, suppression pool and GDCS piping before being returned to the core. Water, therefore, circulates in the PCV effectively to cool down the core.

The relation between pressure inside the PCV and temperature of the suppression pool is shown in Fig.5. For the ordinary PCV, the PCV pressure is the sum of non-condensed gas pressure and steam saturated pressure. On the other hand, for the SSBWR, PCV pressure is equal to the larger of the non-condensed gas pressure and steam pressure because non-condensed gas is evacuated to the wet well. The PCV pressure of the SSBWR, therefore, is less than that of the ordinary PCV. The thin PCV wall cuts construction cost without compromising safety.

The thermal hydraulic behavior in a small LOCA, a 100% break of CUW piping attached to the RPV, was calculated using SAFER model [1]. Internal S/Gs were assumed not to be activated and the ANS + 20% decay heat curve was used for severe analytical conditions. The water level, pressure and temperature during accidents are shown in Fig.6. Here, water levels in RPV and S/P are the height from the top of the core and from the bottom of the S/P. The ADS was activated by the signal of the lower level of water and the RPV was depressurized. The GDCS was activated after ～140 s and the two-phase level in the RPV increased. During the transient, the two-phase level is higher than the top of core and exposure of the core does not occur, and therefore heat-up of the fuel cladding also does not occur. After ～2h, the heat pipe can be activated to transfer heat in the PCV to the outer space. The temperature and pressure of the suppression pool are maintained at 128℃ and 0.25MPa. After 25h, generated and removed heats are balanced and decay heat can be removed with an infinite grace period.

The temperature of the suppression pool is limited to 77°C℃ for a BWR to keep condensation capability of the pool. But for the SSBWR, most of the steam is not condensed in the suppression pool, but on the outer wall of the heat pipe at the final stage of the accident. Non-condensed gas initially fills the dry well and later is confined into the wet well. Therefore the pressure of the non-condensed gas is not increased with a constant ratio of the volumes of the dry well and wet well. The volumes of the suppression pool, dry well and wet well, therefore, can be reduced and a compact PCV can be realized.

5.2. Thermal Hydraulics at the Severe Accident

Even if the isolation valve of the GDCS is not activated, the water level in the RPV can be maintained by condensation of internal S/Gs. But, if both the isolation valves of the GDCS and internal S/Gs are not activated, the water level is gradually lowered in the RPV and core might be exposed. In this case, the core can melt down because of insufficient cooling. Core debris is piled up on the lower inner wall of the RPV and it heats up the RPV wall so that the debris might penetrate the RPV and flow out. To prevent this situation, fused valves connected to the suppression pool by the Direct Vessel Cooling System (DVCS) pipes are installed in contact with the RPV lower wall. Even if there is core melt down, the fused valves are opened by the temperature rise of the RPV heated by the melted debris and the lower part of the RPV is filled with water without any damage. Steam generated at the lower dry well surrounding the lower part of RPV circulates through the dry well, DVCS pipe and lower dry well. Condensed water in the dry well is drained into the lower dry well. Water, therefore, circulates in the PCV effectively to cool down the core for the severe accident as well.

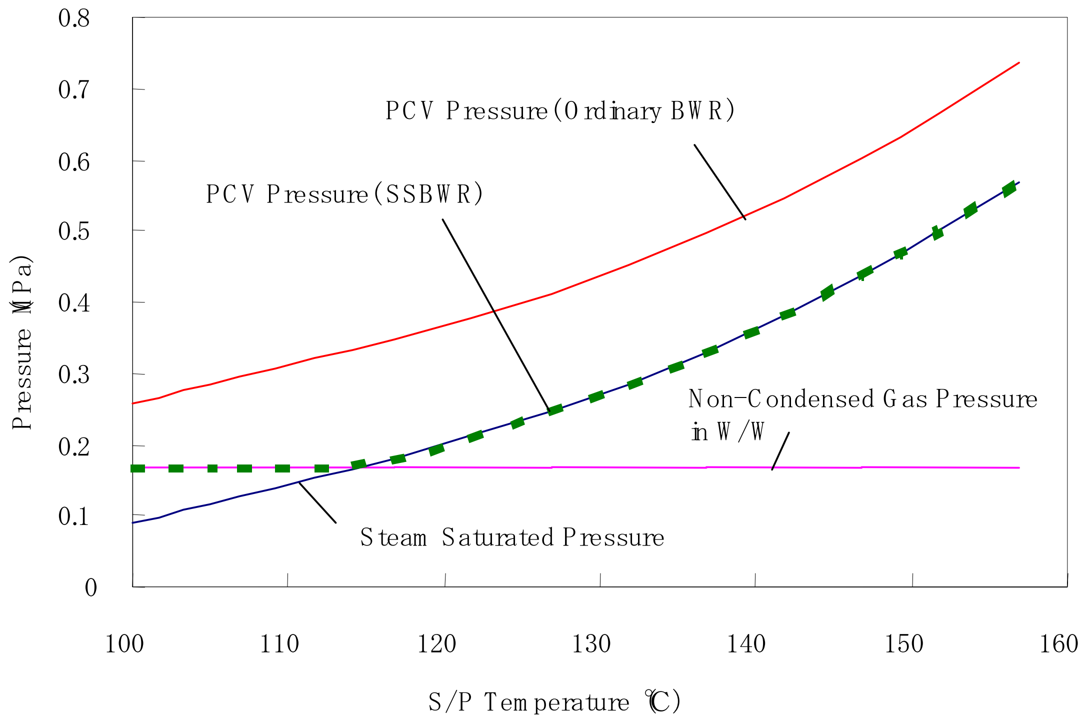
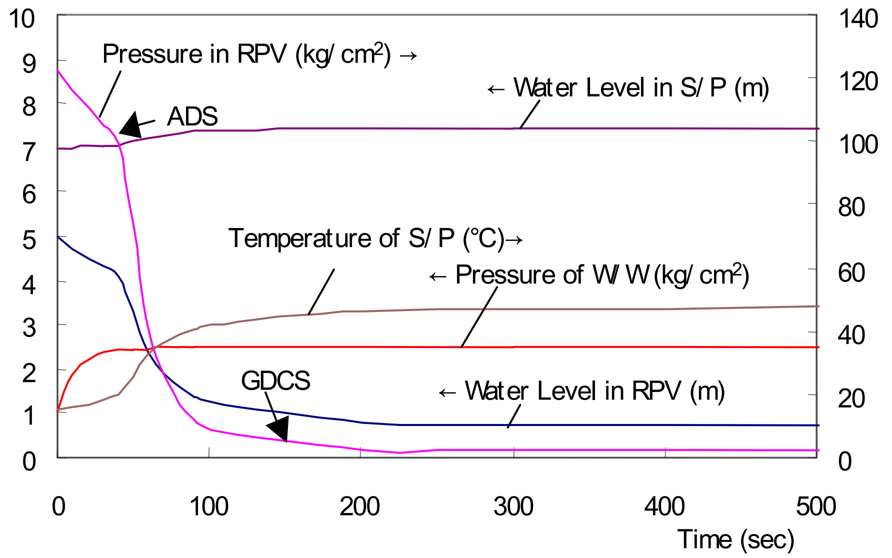
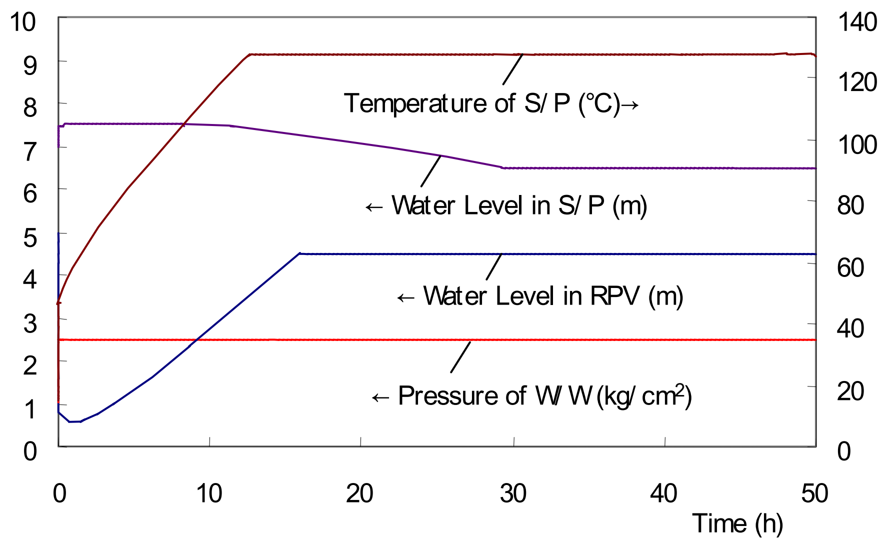


FIG. 5. PCV Pressure



(a) Short Term Characteristics



(b) Long Term Characteristics

FIG. 6. Transient After CUW Line Break

6. PLANT SYSTEMS

Table 2 compares reactor facilities of the SSBWR and ABWR. The primary cooling system is simplified by eliminating steam separator/dryer system and recirculation pumps. Additional internal S/Gs are inserted in the RPV, but the BOP system is simplified due to reduced seismic and shield design. ECCSs are simplified by eliminating the high pressure core flooder (HPCF) and the low pressure flooder (LPFL) with pumps and adding ADS and GDCS without active components. The decay heat removal system is the secondary cooling system of internal S/Gs and the CUW system in cooperation with the RHR (Residual Heat Removal) system.

TABLE I. REACTOR FACILITIES

	SSBWR	ABWR
・NSSS		
Primary Pump	—	10 RIPs
Separator	—	349 units
Dryer	—	6 banks, 22 units
S/G	4 units (internal)	—
CRD	7	205
・BOP System (shield)	Reduced	Shielded
・Auxiliary System		
RHR	— (secondary loop + CUW)	3 systems
CUW	1 system	2 systems
・ECCS	—	RCIC
	—	HPCF
	GDCS+ADS	LPFL+ADS

Driving electric power is very small and the load rate on site is lowered and negligibly small because active components are deleted. By using the indirect cycle, an additional heat exchanger is not always necessary for the third cooling system. The capital costs of construction are reduced by eliminating active components and systems can be contained in a small PCV and reactor building.

CONCLUSIONS

An innovative concept of the safe and simplified boiling water reactor SSBWR has been developed. The reactor has a small size of 484MWth and 150 MWe for multi-purpose distributed energy sources for remote areas and cities. Design targets of the SSBWRs were set as follows: (a) simplicity of operation and maintenance, (b) multi-purpose energy supply, (c) passive safety, (d) cost reduction. A representative concept of SSBWRs provides characteristics of a 20-year core life, an indirect cycle and a passive core safety system with infinite grace period. Operability and maintainability can be largely improved by a super-long life core using heavy water and by reduction of the number of active components such as pumps. An indirect cycle with high temperature secondary steam is used for a multi-purpose energy supply with compact S/Gs of the condensation type in the RPV. Natural coolant recirculation and gravitationally induced safety systems improve the reliability of plants with complete passive safety. ECCSs are simplified by eliminating active systems such as the HPCF and LPFL with pumps, but adding passive systems of ADS and GDCS without pumps. The construction cost can be reduced by compact design of the reactor pressure vessel (RPV), the primary containment vessel (PCV) and the simplified BOP system using the indirect cycle. The reactor contributes to the prevention of nuclear proliferation due to 20-year core life using enriched UO_2 and RPV renewal with no exchange of fuel assemblies. The reduction of waste can also be achieved by having high burnup fuel.

In the future, various levels of needs for energy utilization, maintainability, safety and cost will be proposed for small reactors. A family of SSBWRs will appear using light/heavy/supercritical water, direct/indirect cycle, and so on to satisfy each specification. Additional innovative concepts will be devised on the basis of the representative conceptual design of the SSBWR in this paper.

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NPP WITH VK-300 BOILING WATER REACTOR FOR POWER AND DISTRICT HEATING GRIDS

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Abstract

Specific requirements for nuclear power units for medium-size power and district heating grids are under consideration. Among the main requirements are the following: matching of power unit output with grid stability, enhanced NPP reliability and safety, competitiveness of the power generated.
The design and main characteristics of theVK-300 reactor facility are considered.
The presentation is focused on the most important design features of VK-300. More attention is given to the specific features of the reactor design relating to:

- the original and efficient scheme of coolant circulation and separation,
- the top placement of CPS drive mechanisms,
- the unique system for reactor core emergency cooling.

Reactor passive safety features are given a special emphasis.
The prospects for developing in the Russian Federation nuclear power units with VK-300 reactor facility is analyzed.

1. INTRODUCTION

The analysis of electric power and heat consumption in the Russian Far East and Siberia shows that these regions (especially those with small power systems) need medium capacity power sources for heat and electric power generation that are capable of rivaling the organic fuel plants.

Medium power nuclear plant units should meet specific requirements. Their power is governed by the demand and should not exceed the limit that ensures stability of the power system in the event of a sudden shutdown of the unit. The power limitation for autonomous power systems is unlikely to permit the use of unit with the power of over 250–350 MW in the mix. Larger capital investments due to power reduction should be compensated by simplified reactor and plant designs, less equipment and higher reliability, as well as smaller maintenance and repair costs with assurance of the highest safety requirements.

A major contributor to higher efficiency of medium nuclear power plants is their dual-purpose application both for electric power and heat generation. A nuclear power plant equipped with a VK-300 reactor facility is intended for small- and medium-size power systems as well as for electricity and heat cogeneration.

The design and main characteristics of a two-unit NPP with VK-300 reactors of SBWR-type being developed by RDIPE are as follows:

Main plant data

Number of reactor units	2
Thermal power, MW	750×2
Heat generation capacity, Gcal/h	400×2
Electric power, MW	
under district heating mode	150×2
under condensation mode	250×2
Steam parameters at the reactor outlet	
– pressure, MPa	6.86
– temperature °C	285
– moisture content %	0.1
Reactor steam output, t/h	1370
Number of hours of using installed power per year	7000
Number of hours of using nominal power for heat generation per year	5600
Uranium load, t	32.2×2
Fuel enrichment, %	4
Fuel burnup, MW·day/kg	42.4
Gross efficiency	
– in district heating mode	0.206
– in condensation mode	0.333

2. VK-300 REACTOR FACILITY

The detailed design of the VK-300 reactor facility to replace a dual-purpose production and power reactor in operation as part of the Krasnoyarsk Nuclear Complex, was started in 1997 after its draft design was completed. The draft design stage has been a vivid demonstration of the fact that it is possible to realize the design of an innovative simplified boiling-water reactor the safety of which may be based on passive principles. The basic emphasis in the design was laid on the achievement of two major goals:

- To build a nuclear power unit for electricity and heat cogeneration predominantly in regions remote from the center of the Russian Federation. Also, this nuclear power unit should have limited power which is typical of the heat and electrical networks in operation (or under construction).
- To build a nuclear power unit characterized by a high safety, reliability and competitiveness level. It is taken into account that the NPP will normally be near the city boundaries or immediately within the precinct of a city, i.e. will have a small controlled area.

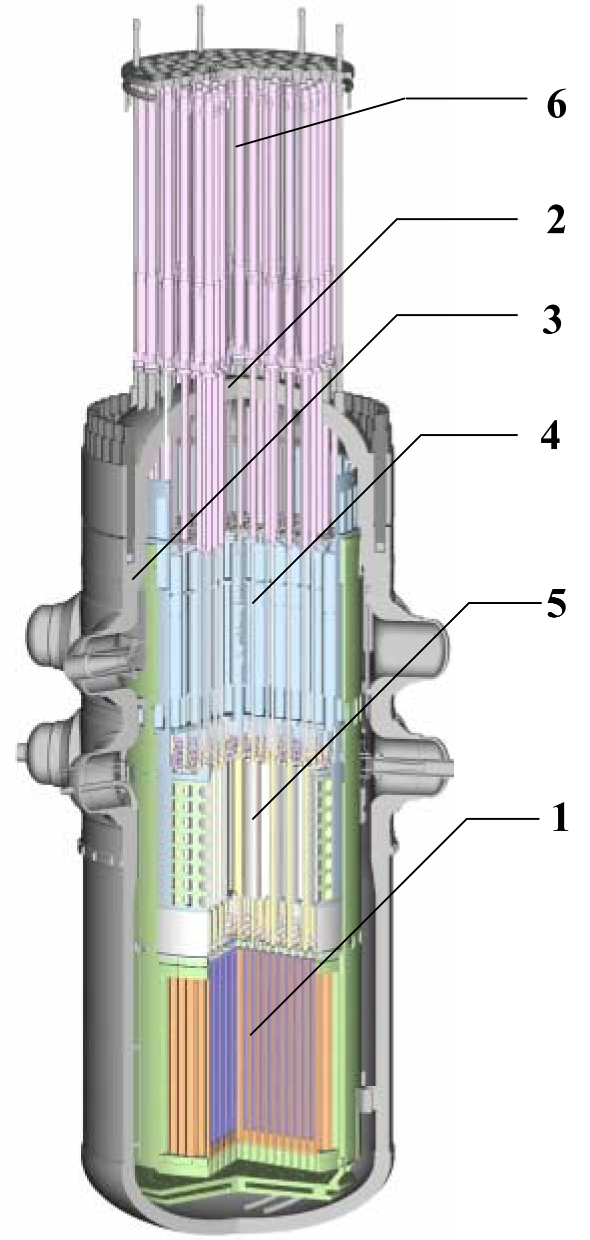
The reactor facility design uses innovative approaches based on previous experience of designing and operating similar nuclear reactors. Thus, a VK-50 reactor operated in the Russian Federation is used as a prototype for the VK-300 reactor. The experience of the design and long-term operation of the boiling-water small-size VK-50 reactor in Dimitrovgrad turned out to be extremely useful in the development of the VK-300 reactor. International achievements in the area of designing and operating boiling-water reactors, especially with respect to the design of passive safety systems, have also been taken into account in the design of VK-300 reactor.

The VK-300 design uses the basic equipment developed and manufactured for reactors of other types. Thus, the VK-300 design uses the WWER-1000 reactor vessel. It is evident that it is difficult, time-consuming and expensive to design and launch into production a new pressure vessel for a power reactor. So using in the design an already completed development of a nuclear reactor vessel considerably facilitates and simplifies the task of designing the VK-300 reactor. Production facilities for manufacturing such pressure vessels have been built in the Russian Federation. Besides, there are really over 10 manufactured vessels and it is an economically expedient task to recover them in national economy.

The VK-300 reactor uses WWER-1000 fuel elements and experimentally optimized cyclone separators that were designed for use in WWER-1000's vertical steam generators.

Therefore, the basic equipment for the innovative boiling-water VK-300 reactor has been well developed and has an extensive experience of operation.

The general view of the VK-300 reactor is given in Fig. 1.



1 – fuel assemblies;
2 – reactor lid;
3 – reactor vessel;
4 – steam separators;
5 – natural circulation guide tubes;
6 – control rod drivers

FIG. 1. Boiling reactor VK-300 assembly.

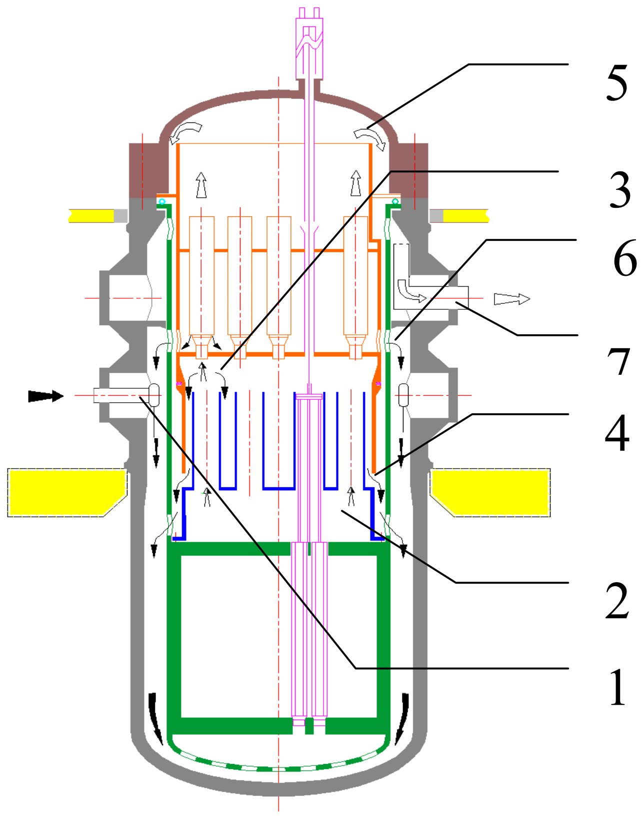
3. ORIGINAL SCHEME OF NATURAL CIRCULATION AND SEPARATION IN THE REACTOR

Much attention in the design of the VK-300 reactor was given to the development of a natural circulation circuit. The designers proceeded from the following mutually exclusive goals:

- To limit the reactor height for ensuring an acceptable layout of the nuclear island with regard for the use of the top placement of CPS drive mechanisms.
- To reach the maximum power in the reactor to improve the economic performance.

To limit the reactor height means to limit the natural circulation pressure and rate which does not permit to increase the power of the core. A simple decision is to raise the steam quality in the core and thus raise the natural circulation pressure in the reactor. But this way leads to deteriorated neutronic and thermal-hydraulic characteristics of the reactor, worse heat engineering reliability, reduced circulation stability and undesirable consequences of an excessive reactivity void effect.

In this connection, the way of reducing the hydraulic resistance of the coolant circulation circuit was chosen. Primarily, an original coolant circulation and multi-stage separation scheme was used.



1 – feed water;
2 – out-core-mixing chamber;
3 – preliminary separation chamber;
4 – pre-separated water outlet;
5 – steam;
6 – major separated water;
7- stream outlet

FIG. 2. Reactor flow diagram.

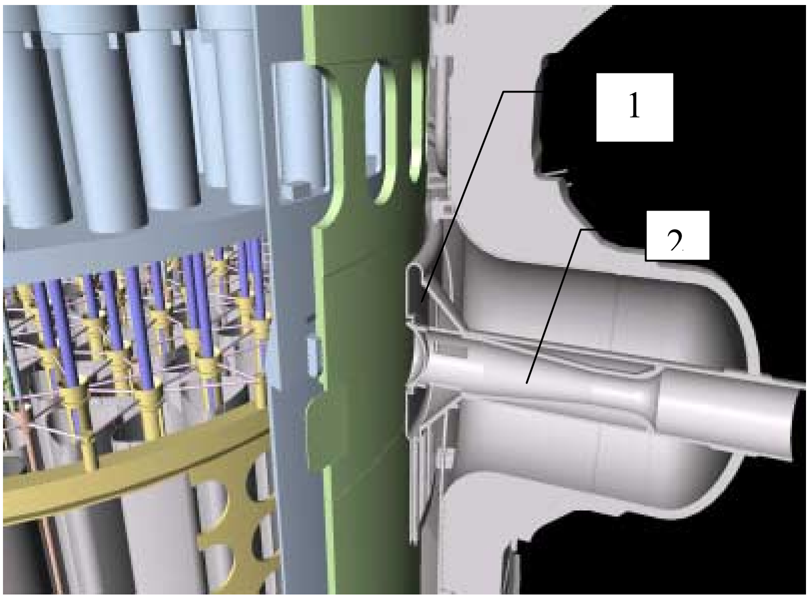
It is generally known that in most modern boiling-water reactors with internal steam separation the entire steam-water mixture flow downstream of the core goes to the separators (normally, of a cyclone type) where moisture is separated. And the hydraulic resistance of the separators turns out to be high. If moisture is preliminary taken from the flow and returned to the core inlet which reduces the mass flow rate of the steam-water mixture through the separators, it is possible to reduce the hydraulic resistance of the circuit and, as a consequence, increase the natural circulation rate. The reactor circulation and separation circuit scheme is shown in Fig.2.

The reactor design includes a unit of draft tubes whose functions are:

- Segregation of the coolant flows going up and down
- Preliminary moisture separation
- Formation of a water inventory (between the draft tubes) that is immediately returned to the core at reactor shutdown or during accidents.
- Creation of a guiding structure for the reactor control rods (which is very important for the upper placement of the CPS drives).

The possibility of the moisture separation after the steam-water mixture leaves the draft tube unit has been proved experimentally.

Structural components of the reactor are shown in Fig.3, 4, 5, 6.



1 – feed water distributor;
2 – safety device (limiter)

FIG. 3. Feed water assembly.

Primary containment of the reactor - an innovative approach to passive safety assurance during accidents with ruptures and loss of heat removal from the reactor

The use of the Primary Containment (PC) is an economic and reliable approach to the safety assurance problem thanks to the use of structurally simple passive safety systems. The PC has a small volume (1 500 cu.m). It performs the functions:

– as a safeguard (additional) reactor vessel;
– as a protective safety barrier that confines radioactive materials within its boundaries during accidents with ruptures of steam, feed water and other pipelines immediately near the reactor;
– of enabling emergency cooling of the reactor with the reactor coolant without the need of an additional coolant inventory.

Located outside the PC are Emergency Cooldown Tanks (ECTs) intended for:

- accumulation of the reactor energy (ensuring the possibility of its transfer to the ultimate heat sink for an unlimited period of time);
- replenishment of the cooling water inventory in the reactor during accidents through the return of the condensed coolant to the reactor.

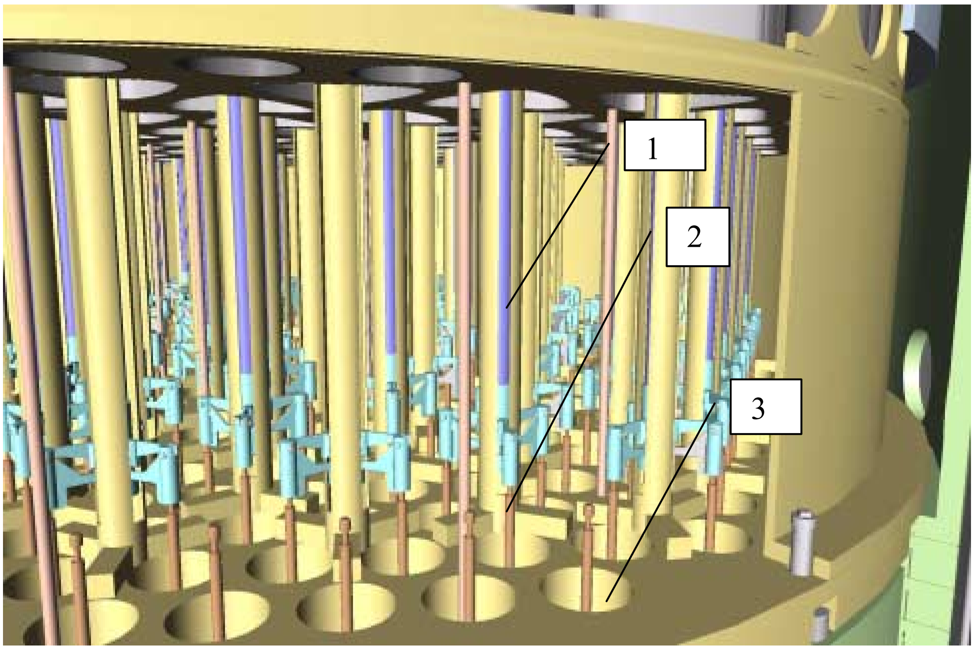
A simplified hydraulic scheme of the PC as a set with the ECTs is shown in Fig.7.

The pressure in the reactor and in the PC is leveled rapidly during ruptures of steam or feed water pipelines inside the PC. It creates conditions for the water inflow from the ECTs to the core via a special pipeline. The initial accident stage proceeds safely without reactor makeup as the water inventory in the reactor is enough to ensure normal heat removal from the core. Later, as the pressure decreases in the reactor and the pressures in the reactor and in the PC are leveled, the water goes to the reactor from the ECTs by gravity. An external circulation circuit is formed - ECT, reactor, PC, ECT. The water from the ECTs is accumulated in the PC with time but it does not affect the serviceability of the tanks as the water inventory therein is enough for filling up the PC volume and successful operation of the “external” natural circulation circuit. It should be noted that the PC is automatically pressurized during accidents with ruptures (using special passive action valves) to exclude releases of the radioactive coolant beyond the PC.

Another class of accidents includes accidents with the loss of heat removal from the reactor due to a turbine failure or accidents in the reactor’s external feed water line. The major task here is to receive heat from the reactor and ensure its normal cooldown. This is ensured by a special system for passive heat removal from the reactor based on the use of steam condensers located in the PC around the reactor. These condensers are connected with the reactor through pipelines that are flooded with the primary circuit water during normal operation of the reactor. When the water level decreases in the reactor, the upper pipeline is opened to let the steam from the reactor to the condensers and the condensate flows back to the reactor. The condensers as such are cooled with water from the emergency cooldown tanks. The system is based on a fully passive principle and intended for natural heat transport from the reactor to the emergency cooldown tanks.

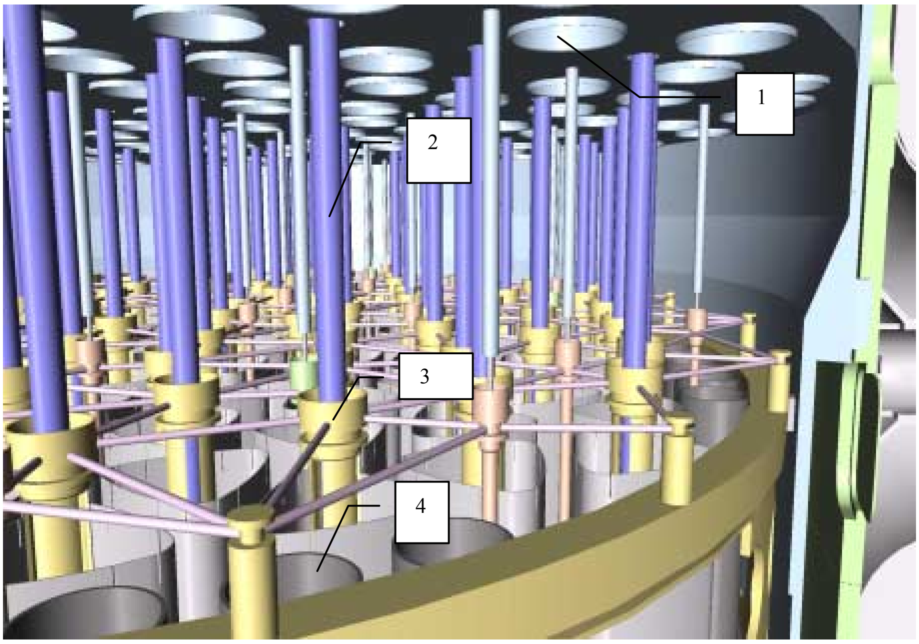
It should be noted that the emergency cooldown tanks are also intended for receiving the discharge from the reactor safety valves located inside the PC (Fig.8).

The above examples show that the heat from the reactor is accumulated in the emergency cooldown tanks. The heat capacity of the tanks as such is enough for independent operation throughout the day (i.e. without personnel interference). This interval may be prolonged for an infinite period of time thanks to the operation of the heat removal system from the tanks to the ultimate heat sink. This is a simple and reliable system consisting of two heat exchangers



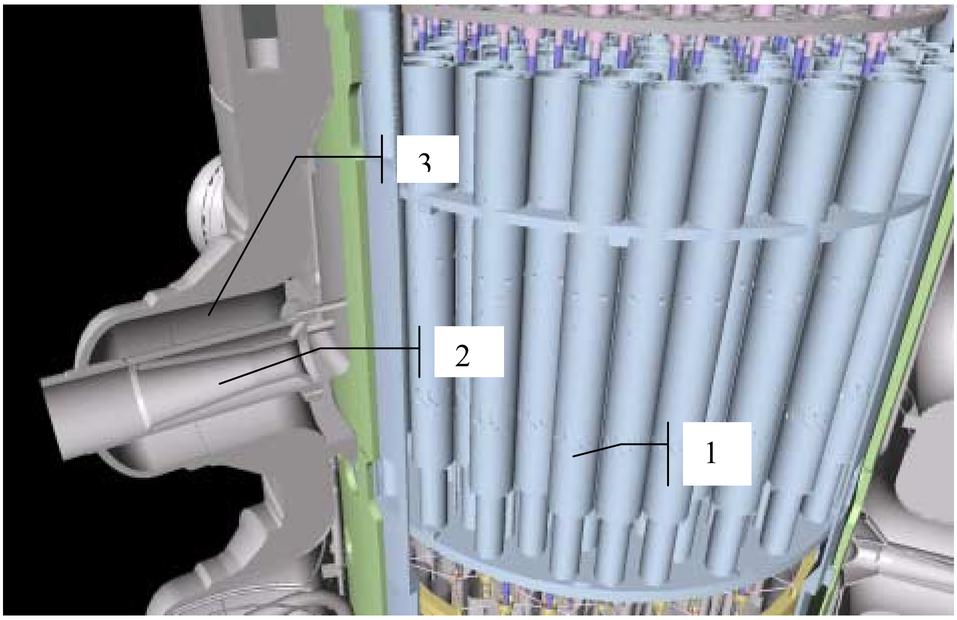
1 – control rod guide;
2 – control rod tale;
3 – fuel assembly outlet

FIG. 4. Out core mixing chamber assembly.



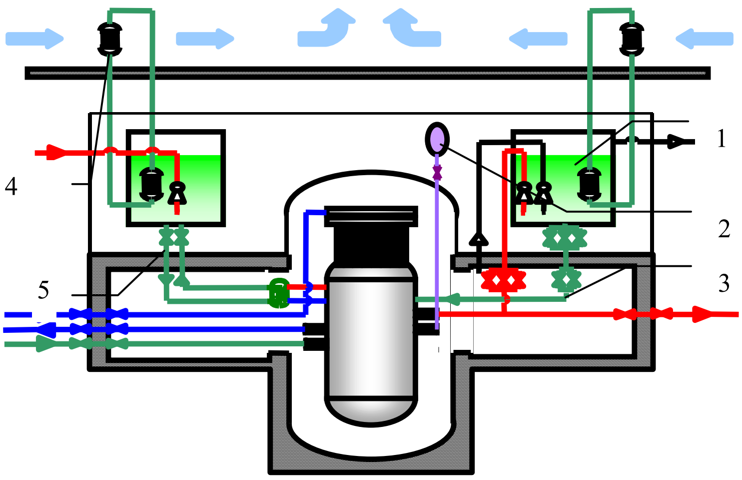
1 - steam separator support plate;
2 - control rod guide;
3 - fixing coordinate mechanism (fixing grid);
4 - guide circulation tube

FIG. 5. Preliminary separation chamber assembly.



1 - steam separator;
2 - steam collection device;
3 - steam outlet

FIG. 6. Steam separator assembly.



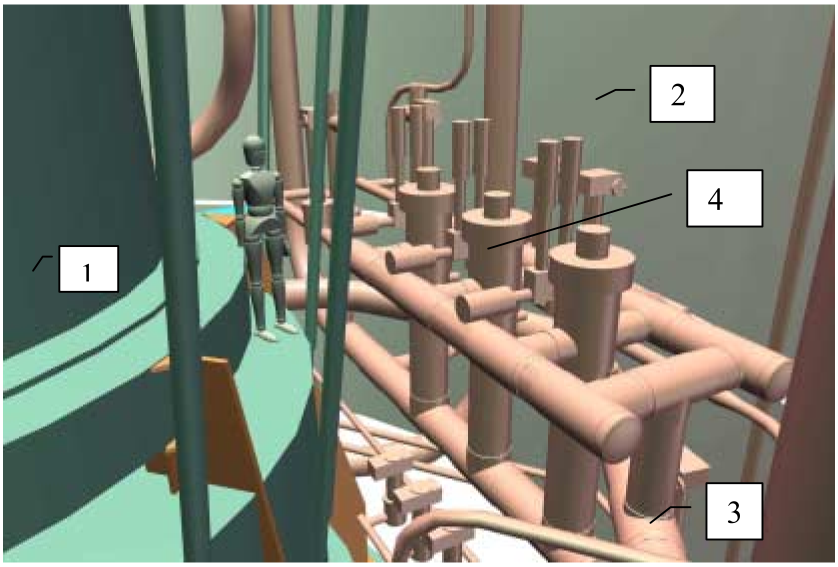
1 - emergency cooling tank;
2 - liquid absorber storage vessel;
3 - emergency core flooding system;
4 - air heat transfer system;
5 - emergency core cooling system;
6 - preliminary protective containment

FIG. 7. Reactor plant flow diagram.

connected with pipelines. One of the heat exchangers is plunged into the emergency cooldown tank water and the other is installed in the atmospheric air flow outside the reactor hall. The coolant in the system is water circulating in the circuit naturally without circulation boosters.

4. CONTROL OF THE FISSION CHAIN REACTION

Reactivity effects and coefficients that set up the basis for the reliable controllability of the reactor and its stable operation are very significant for the successful performance of control function of the fission chain reaction. The VK-300 reactor has just a small reactivity margin for burnup thanks to partial refueling and the use of the burnable poison. The minimization of the reactivity margin ensures prerequisites for designing a simplified CPS system with “light” rods that mitigates the consequences of an accident with self-withdrawal of the CPS rods.



1 – reactor thermal insulation;
2 – wall of the preliminary protective containment;
3 – main steam guide tube;
4 – relief valve

FIG. 8. Reactor relief valve assembly.

The reactor has two reactivity control systems using different principles of operation. The first of the systems is a traditional rod system including 90 drives of the CPS actuators. Each of the drives simultaneously moves the control rods in the three adjoining core fuel assemblies.

The second reactivity control system is a liquid system intended for introducing the boric acid solution to the reactor coolant during accidents or at failures of the reactivity control rod system. The system consists of pressurized hydroaccumulators with the boric acid solution.

The analysis has proved that the VK-300 reactor facility is highly safe. The probability of a severe damage to the core does not exceed $1{,}4 \cdot 10^{-7}$.

5. VK-300 RF DEVELOPMENT STATUS

The VK-300 RF detailed design was completed in 2000. Works have been planned and are underway for the experimental substantiation of the reactor circulation circuit as required to reduce excessive conservatism that was laid in the reactor calculations. Additionally, experimental and design work is required for the CPS rod drives for which purpose full-scale drives, bars and CPS rod coupling devices are manufactured to be used in bench tests. The entire R&D complex is proposed to be completed during 2-3 years after which the reactor contractor design with the R&D results taken into account will be issued.

CONCLUSION

The VK-300 design was developed as a substitute power facility for the Krasnoyarsk nuclear complex. Good economic performance was achieved during the design in satisfying high safety requirements not only for the VK-300 reactor facility but also for the nuclear cogeneration plant of the Krasnoyarsk nuclear complex. The achieved result has enabled to start to consider the problem of using the design in the Russian Federation's other sites in the regions short of electric power and heat.

DEVELOPMENT OF ENABLING TECHNOLOGIES FOR THE INDIAN ADVANCED HEAVY WATER REACTOR

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Abstract

The Indian Advanced Heavy Water Reactor (AHWR) is a vertical pressure tube type boiling light water cooled and heavy water moderated reactor with a nominal power of 235 MWe. The reactor is fuelled with Uranium-233-Thorium MOX together with Plutonium - Thorium MOX, with the former producing a major fraction of power. The AHWR incorporates a number of enabling technologies, of importance for advanced reactors in general, and next generation pressure tube type reactors in particular. The reactor achieves a slightly negative void coefficient of reactivity. The advanced design of the AHWR coolant channel facilitates its easy replaceability. During power operation the reactor is cooled with natural circulation. The technological issues inherent with a naturally circulating two phase system are being solved through analytical as well as experimental programmes. Other passive systems provided in the reactor include passive decay heat removal system, passive containment cooling system and a passive containment isolation system. The reactor incorporates features to achieve enhanced economic performance through elimination of some safety grade expensive equipment, and an efficient management of energy produced in the core. At present the reactor is in a detailed design and development stage. Apart from the design of the reactor, studies on its thorium based fuel cycle are also being carried out. Some results of the afore-mentioned work are presented in the paper.

1. BACKGROUND

The Indian Advanced Heavy Water Reactor (AHWR) is a vertical pressure tube type, boiling light water cooled and heavy water moderated reactor (Fig. 1, next page) with a nominal power of 235 MWe. The reactor is fuelled with ^{233}U - Th mixed oxide (MOX) together with Pu-Th MOX, with the former producing a major fraction of power. AHWR is nearly self-sustaining in ^{233}U. A small external feed of plutonium is required to make the reactor critical. At present the reactor is in a detailed design and development stage. Some important data of the reactor is given in Table I.

Several features of the AHWR are associated with specific enabling technologies, particularly relevant for next generation reactor systems and fuel cycles. Some of these features are discussed in this paper.

TABLE I. IMPORTANT DESIGN PARAMETERS OF AHWR

Reactor power	750 MW_{th}	Coolant inlet temperature	544 K (nominal)
Core configuration	Vertical, pressure tube type	Total core coolant flow rate	2576 kg/s
Fuel	$(Th-^{233}U)O_2$ -30 pins and $(Th-Pu)O_2$ -24 pins	Steam generation rate	362 kg/s
Central absorber	^{164}Dy -2.5%	Steam drum pressure	7 MPa
Enrichment, ^{233}U, Pu	3.45, 3.0 wt %	PHT loop height	39 m
Coolant channels	452	Calandria diameter	8000 mm
Pressure tube ID	120 mm	Calandria height	5000 mm
Lattice pitch	270 mm (square)	Heated fuel length	3.5 m
Moderator	D_2O + Void Tube	Average heat rating	8.8 kW/m

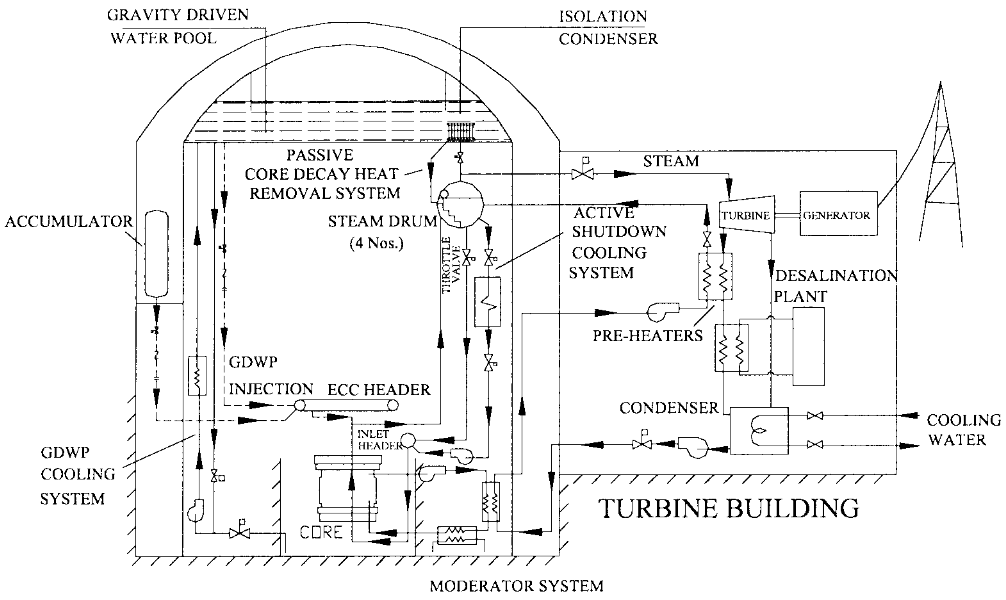


FIG. 1. Schematic Diagram of Advanced Heavy Water Reactor.

2. REACTOR PHYSICS AND FUEL DESIGN

The reactor has a slightly negative void coefficient of reactivity together with an efficient fuel performance. This has been achieved by optimising the fuel cluster, and by incorporating new features in the lattice design like tight lattice pitch, inter-lattice void tubes and a displacer rod in fuel cluster. The latter also serves as a conduit for Emergency Core Cooling System (ECCS) water injection.

A unique composite MOX fuel cluster [1] is used to obtain a higher fraction of power from thorium. The cluster has three circular arrays of fuel pins (Fig. 2) in which (Th, Pu) MOX pins are located in the outermost array, where neutron flux is high, to optimise the fuel inventory and to effect a reduction in void reactivity. A slightly under-moderated system results in a harder neutron spectrum, which contributes to the negative component of void reactivity. Further reduction in moderator inventory, and hence in void reactivity, is obtained by incorporating void tubes in inter-lattice locations. A small quantity of slow burning poison, dysprosium, is used in the central displacer region of cluster to turn the void reactivity negative. Finally, a relatively low value of lattice pitch of 270 mm is needed to help attain a negative void reactivity, with a low burden of this burnable poison. The increase in absorption in dysprosium, upon voiding, is responsible for negative void reactivity. This characteristic leads to the void coefficient remaining slightly negative at all values of fuel burn-up. The zirconium oxide displacer region created in the center of cluster reduces pin power peaking within the cluster. Differential enrichment of fuel in the cluster also improves the power distribution in the cluster and fuel burn-up. A large number of relatively low diameter fuel pins are used for adequate heat extraction/removal by natural circulating coolant and to improve fuel utilisation.

ECCS water is injected directly in the core region, thus providing an efficient mode of cooling the fuel assembly. ECCS water from accumulators and GDWP enters the central tube (displacer rod) of the fuel assembly. Water is sprayed on fuel pins through the holes provided in the central tube of the fuel assembly. The effectiveness of this scheme has been assessed in the first phase of the experimental programme by spraying water on cold rods of a simulated set-up.

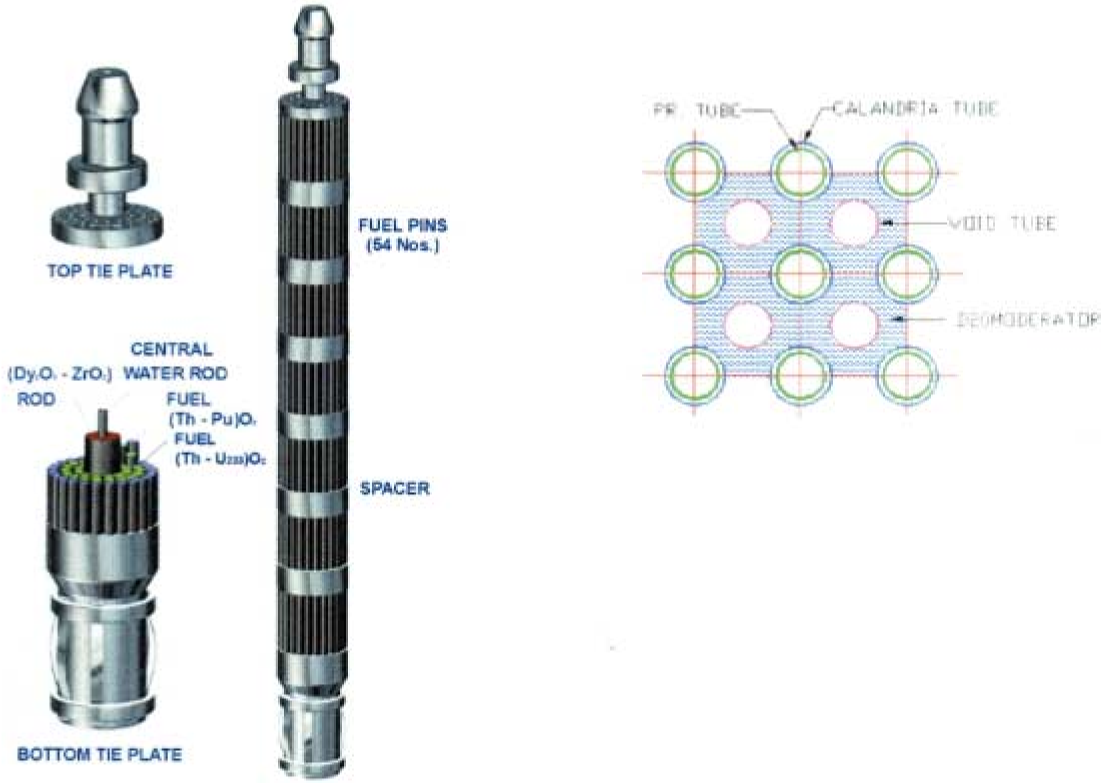


FIG. 2. AHWR Fuel cluster and lattice layout.

The radial power distribution is flattened to improve thermal margins and fuel utilisation. A multi-zone fuelling scheme has been worked out to achieve it. The average fuel exit burn up is 24 MWd/kg. Linear stability analysis is carried out to ascertain the spatial stability of the core. A large number of in-core self-powered neutron detectors are provided for core monitoring and flux mapping. A quadrant power distribution control system is employed to control the spatial instabilities. A large negative value of void coefficient is avoided to reduce the positive reactivity realised upon void collapse, and to maintain a uniform axial flux shape. This also avoids a bottom-peaked axial flux shape which is undesirable as the control devices in AHWR enter the core from top. In addition, the resulting distorted axial flux shape could decrease spatial stability of the core and adversely affect fuel utilisation. A coupled neutronics and thermal hydraulics code is under development for spatial stability analysis of the AHWR core in time domain. Twelve adjuster rods, along with boron in moderator, are used for control purposes. Two independent shutdown systems, one based on mechanical shut-off rods falling under gravity and the other based on liquid poison injection in the moderator, are provided for reactor shutdown. ^{233}U is used in near self-sustaining mode in a closed fuel cycle in AHWR and recovered ^{232}Th too is proposed to be recycled. A significant amount of ^{232}U, ^{228}Th and ^{229}Th is expected in fuel. The WIMS library is being updated to include relevant isotopes [2]. A preliminary estimate of the ^{232}U inventory in fuel has been made by WIMSD code. A linkage between WIMSD and ORIGEN required for a better estimate of ^{232}U and other actinides is under development. Some results of core physics calculations are shown in Fig. 3(a) and Fig. 3(b). The computer codes and nuclear data used for theoretical calculations will be validated by comparing the results with experiments to be performed in a critical facility.

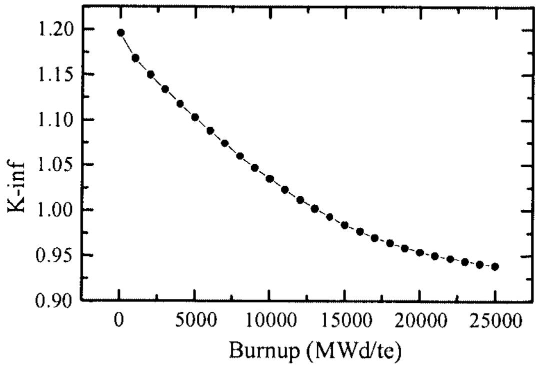


FIG. 3(a). K-inf as a function of burnup in AHWR.

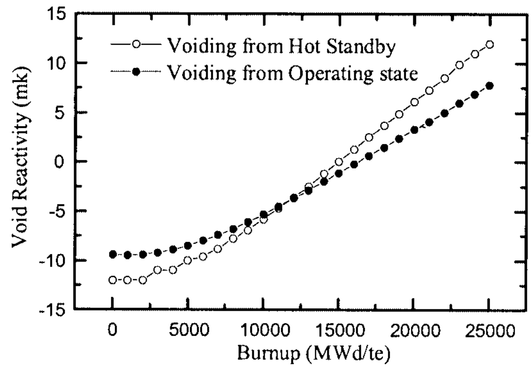


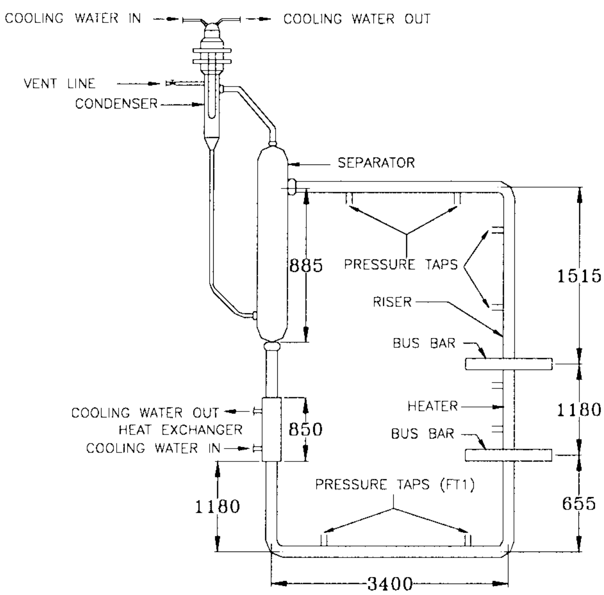
FIG. 3(b). Coolant void reactivity as a function of burnup in AHWR.

3. NATURAL CIRCULATION BASED HEAT REMOVAL

Since natural circulation is the mode for heat removal from the core of the AHWR, it is necessary to examine the possibility of occurrence of various types of instabilities which may further get coupled with the neutronics to induce power oscillations. It is essential to predict the stable and unstable zones of the reactor operation during the design stage so that methods of suppressing or procedures to avoid instabilities can be accordingly worked out. For this purpose, theoretical models were developed by solving the conservation equations of mass, momentum and energy applicable to homogeneous equilibrium flow based on linear stability theory. Comprehensive models for the neutron kinetics (which include a point kinetics model for in-phase mode oscillation and a coupled multi-point kinetics model or modal point kinetics model for out-of-phase mode oscillation) and thermal dynamics of the fuel are incorporated into the above model to investigate the coupled neutronic-thermohydraulic instabilities. To validate the theoretical model, experiments are being conducted in a two-phase natural circulation loop under different operating conditions. The experimental loop (see Fig. 4) consists of a vertical tubular heater directly heated by electric current up to a maximum power of 80 kW. The ID of heater is 52.5 mm. The subcooled water enters the heater at the bottom and gets heated as it rises through the test section. The steam-water mixture coming out of the heater rises through the riser section and is passed on to a vertical separator. The steam then goes to the condenser where it gets condensed and the condensate is returned back to the separator through a pipe that joins the separator at the bottom. The condensed steam and water mixture then flows down through a double pipe heat exchanger. The purpose of this heat exchanger is to maintain the desired inlet subcooling. Fig. 5 shows a comparison of the measured natural circulation flow rate with the analytical predictions at different pressures. It can be observed that the analytical model closely simulates the natural circulation behaviour of the loop.

One of the important types of instability is the Ledinegg instability which is a static type instability encountered due to the negative slope of flow vs pressure drop curve of the system. Figure 6 shows the comparison of Ledinegg type instability maps at different pressures. It can be observed that the instability decreases with increase in pressure. This may be due to the fact that with an increase in pressure the void fraction decreases for the same quality significantly, which can reduce the S-shaped variation of the irreversible losses (i.e. $\partial \Delta p_f / \partial w$, where Δp is the friction pressure drop and w is the mass flow rate) responsible for the occurrence of the Ledinegg type instability. An interesting observation is that this instability almost vanishes for the AHWR (i.e. shifts beyond the operating envelope of power for the AHWR) when the operating pressure is more than 0.7 MPa. One simple way of completely avoiding this instability for the AHWR is by maintaining subcooling less than 10 K, as observed from the figure.

During the in-phase mode oscillations, the flow oscillations among the channels, along with the external loop (i.e. down-comers), occur without any phase difference among them. During the out-of-phase mode oscillations, the flow oscillations among the channels occur with some phase difference without any oscillations taking place in the external loop. Analysis indicates that the out-of-phase mode oscillations are more dominant as compared to the in-phase mode oscillations in the reactor (since the former is having less stable area than the latter) because of the extra single-phase friction in the down-comers which stabilises the in-phase mode oscillations.



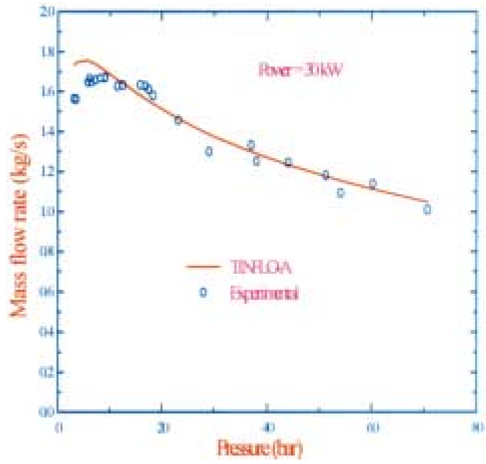


FIG. 4: A simple schematic of high-pressure natural loop.

FIG. 5: Comparison of measured and circulation redicted mass flow rates.

One of the important parameters to indicate the stability margin is the decay ratio (DR) which is defined as the ratio of two successive amplitudes of oscillations. The contour lines of the constant decay ratio for in-phase mode of oscillations at different powers and inlet subcooling are shown in Fig 7 [3]. Also, the constant feed water temperature (t_{feed}) lines along with the constant channel exit quality (x) lines are shown in the same figure. These maps are useful for design of the reactor because they give an indication of the stability margin under various operating conditions. From these maps it is clear that it is possible to have decay ratio less than 0.4 for subcooling less than 10 K, which implies that the reactor can have sufficient stability margin for operating conditions at the above subcooling. With an increase in channel exit quality at a particular subcooling, the DR increases. At a particular power, with increase of subcooling at core inlet the DR increases. A decrease of feed water temperature at a constant power increases the subcooling, which has a destabilising effect.

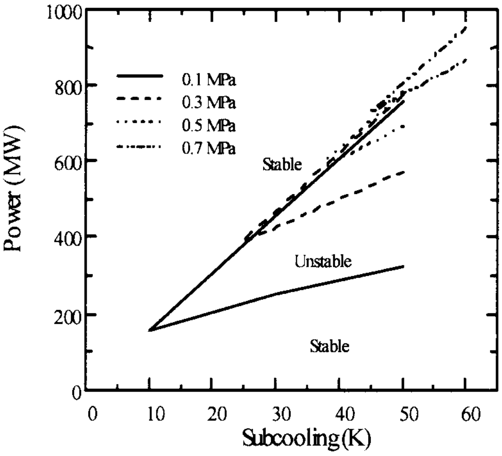


FIG. 6. Effect of pressure on Ledinegg type instability.

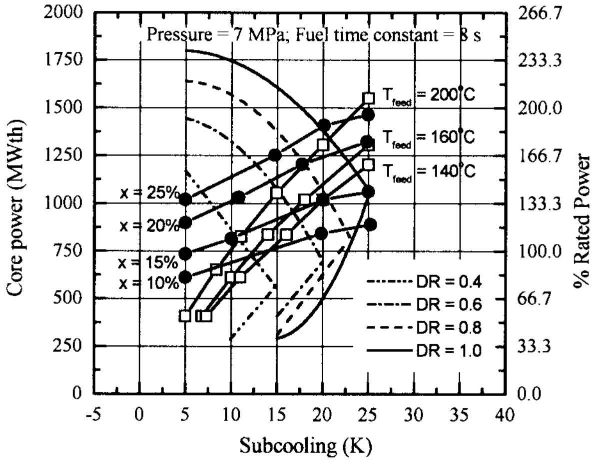


FIG. 7. Decay ratio map of AHWR for in-phase mode of oscillation.

Core decay heat is also removed by natural circulation through Isolation Condensers (IC) during reactor shutdown. Steam formed due to decay heat passes through the IC tube bundles which are submerged in a large pool of water, called Gravity Driven Water Pool (GDWP), located high above the core in the dome region of the containment. The steam condenses inside the IC tubes and the condensate returns to the steam drum by gravity. The computer codes developed for the analysis of natural circulation and its stabilities will be further validated against data to be generated in an Integral Test Loop (ITL).

4. PASSIVE CONTAINMENT COOLING

A passive containment cooling system (PCCS) has been provided for the long term cooling of the reactor containment following a postulated Loss of Coolant Accident (LOCA). Two alternative designs for the passive cooling of containment are under consideration. In the first alternative, the PCCS removes energy released into the containment through immersed condensers kept inside GDWP (Fig. 8). In this case, steam condenses inside the tubes of the immersed condensers and the non-condensables are periodically purged. An important aspect of the functioning of this system is the degradation of the heat transfer coefficient in vertical tubes of immersed condensers due to the presence of non-condensables. In the second alternative, the cooling coil of the Passive Containment Cooler (PCC) is connected to a water pool above it. Containment steam condenses on the outer surface of the tubes. Water from the pool circulates through the tubes by natural circulation.

Experiments to study the system response behaviour have been conducted on a small-scale model of the PCCS (first alternative). The volume scaling of the set-up is approximately 1:3000. The experimental results proved the efficacy of the system [4]. An important aspect of the functioning of this system is the degradation of heat transfer coefficient in vertical tubes of immersed condensers (ICs) due to the presence of noncondensables. Separate effect tests on the full-scale tubes of IC have been conducted to study the effect of noncondensable gas on steam condensation [5]. Some results are shown in Fig. 9.

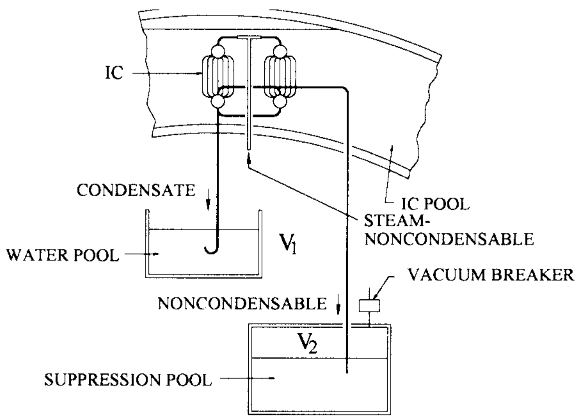


FIG. 8. Passive containment cooling system with immersed condensers.

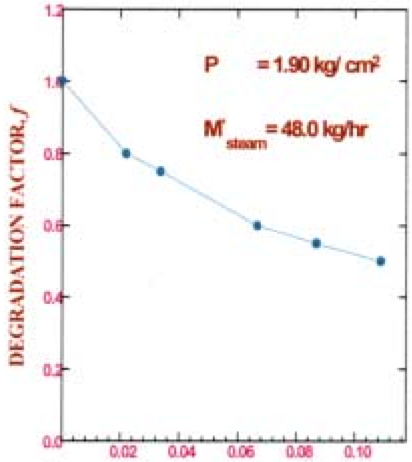


FIG. 9. Variation of degradation factor with air mass fraction.

5. PASSIVE CONTAINMENT ISOLATION

To achieve isolation of containment, following LOCA, in a passive manner, the reactor building air supply and exhaust ducts are shaped in the form of U-bends of sufficient height. In the event of a postulated LOCA, a part of the containment pressurises due to the release of steam. This pressure rise is utilised for swift establishment of siphons to pour GDWP water into ventilation duct U-bends. When appropriately filled, these U-bends act as water seals between the containment and the external environment. These devices are provided in addition to conventional isolation features.

6. COOLANT CHANNELS

Each of the 452 coolant channels of AHWR consists of a Zr-2.5Nb pressure tube extended in either direction by means of stainless steel top and bottom end fittings respectively. The bottom end fittings are coupled to the feeders through high-pressure high temperature couplings and top end fittings are welded to tail pipes. The design facilitates speedy removal and replacement of pressure tubes without disturbing any other permanent part of the reactor. The guiding philosophies for design of the coolant channel are given below:

a) On power fuelling
b) Annulus gas sampling for meeting Leak Before Break (LBB) criteria
c) Feasibility for In Service Inspection (ISI):
d) Easy installability and replaceability of pressure tubes
e) Easy maintainability
f) Direct injection of ECCS water to fuel pins in the event of LOCA
g) Minimisation of pressure drop to promote natural circulation.

The pressure tube, end fittings and a portion of the tail pipe will be shop-assembled and installed as a single sub-assembly. This reduces site work and construction time, and offers better quality control on the assembly. As the design life of the pressure tubes is lower than that of the remaining components, replacement of pressure tubes will be required during the lifetime of the reactor. The technology for easy replacement of the pressure tube has been developed. The design philosophy for easy replaceability incorporates the following: reduction in the number of components to be removed and replaced, use of simple and proven technologies for replacement, minimum interference with the neighbouring channels, and minimum refurbishment of remaining components.

7. STUDIES RELATED TO THORIUM BASED FUEL CYCLE TECHNOLOGIES

Apart from reactor fuel design, studies on different aspects of the thorium fuel cycle are being carried out. One of the main hurdles in thorium fuel reprocessing is the highly stable nature of thorium dioxide, which makes its dissolution more complicated than uranium dioxide. In addition, the thorium fuel cycle has to take care of the radiological problems posed by the presence of ^{232}U in separated ^{233}U and that of ^{228}Th and ^{229}Th in separated ^{232}Th affecting its re-fabrication.

The reprocessing processes until now have been concerned essentially with the extraction and utilisation of fissile materials. In the thorium-uranium fuel cycle to be adopted for AHWR, the extraction and utilisation of fertile thorium is also being planned. THOREX (Thorium-^{233}U Extraction) process has not matured to the same extent as PUREX (Plutonium-Uranium Extraction) process. Some aspects of THOREX process need special attention namely – difficulty in dissolution of irradiated thoria, longer cooling period for complete recovery of ^{233}U and handling problems of ^{233}U and separated thorium due to the presence of ^{232}U, ^{228}Th and ^{229}Th.

7.1. Dissolution of Irradiated thorium fuel

A small amount of fluoride addition to nitric acid is required for the dissolution of the more inert thorium dioxide. The use of fluoride, however, enhances the corrosion of stainless steel (SS) equipment. This problem is mitigated by the addition of aluminium nitrate to complex the free fluoride ion during dissolution. During the dissolution of thorium fuel, dissolution of the Zircaloy clad also takes place to a small extent. Parametric dissolution studies were done at a laboratory scale to establish the lowest acidity and fluoride ion concentration for dissolving thoria fuel at an acceptable rate with a view to minimising the feed adjustment for subsequent solvent extraction step in THOREX process. Studies on the corrosion rates of SS and Zircaloy in dissolving mixtures were carried out simultaneously. Comparison was made between HF and NaF as an agent for fluoride ions. The study revealed dissolution rates with NaF were marginally higher initially compared to HF but the time taken for quantitative dissolution in both the cases remained the same.

The addition of MgO in fuel during fabrication as a sintering aid has been found to improve the dissolution rate [6]. Dissolution studies have been carried out with MgO doped pellets with doping ranging from 0.5%- 2.5% using HNO_3 and HF, and 1.5% has been found to be the optimum value. The dissolution time has reduced almost by an order of magnitude compared to that of a non-doped pellet.

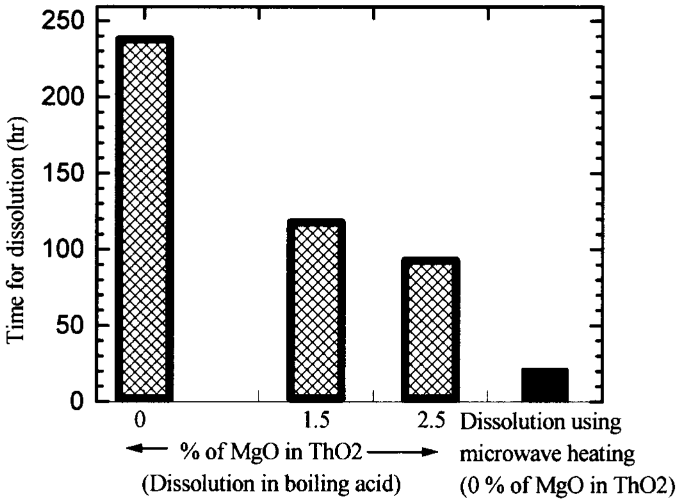


FIG. 10. Dissolution of thoria pellets doped with magnesia.

In addition, studies have also shown an improvement in dissolution of thoria pellets without crushing them to powder, using a microwave heating technique. Experiments on a lab scale were conducted in pressure vessels in nitric acid. The studies have shown a remarkable difference in the rate of dissolution if HF is replaced with NaF. An elaborate study was also carried out to observe the leach rate of Zircaloy and SS coupons under the best condition for thoria dissolution with HNO_3 and NaF. This process is more amenable for recovery of thorium wastes in fabrication plant. For reprocessing the throughput is very high and more R&D is needed for the development of high power microwave sources.

7.2. Extraction process

The versatile extractant TBP in hydrocarbon diluent still remains the best choice for the extraction of both ^{233}U and thorium or for the selective extraction of ^{233}U alone. The TBP content in the diluent (usually Shell Sol-T, dodecane or n-paraffin) varies depending on the final product required.

Alternate organic extractants are also being considered for the thorium extraction process. Tri-iso-amyl phosphate is reported to have higher solubility for extracted plutonium and thorium. Amides are also being investigated as potential candidates for selective extraction of uranium based on its superior uranium-thorium separation factor. Further studies are required to evaluate its chemical and radiolytic degradation as well as third phase formation.

In India, experience of 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 ^{233}U/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.

7.3. Thorium fuel fabrication

Presently thorium dioxide bundles loaded in PHWRs for initial flux flattening are being fabricated using a conventional powder-pellet route. This large-scale fabrication experience has given an insight to various aspects of thorium fuel fabrication. The radiological problems, mentioned earlier, necessitate remote fabrication in shielded facilities. Fabrication processes like Sol-Gel process, Advanced agglomeration technique and Pellet impregnation technique, which are more suitable for remote fabrication, are being developed.

7.4. Fuel cycle facility for AHWR

AHWR is designed for a closed fuel cycle necessitating reprocessing of irradiated fuel and refabrication from the recovered material. A fuel cycle facility is being proposed for AHWR. The activity of separated ^{233}U increases with time due to the presence of ^{232}U in ^{233}U. This allows only few days time from the re-processing plant to completion of fuel pin fabrication. It is also proposed to use the recovered thorium from the reprocessing plant in the fuel fabrication without allowing much time delay. In this case, the ^{228}Th present in ^{232}Th will pose radiation problems. Hence, the reprocessing & fabrication plants will be co-located.

It is proposed to continue with fabrication development activities of all the above mentioned processes to a level that can be used for AHWR fuel production. However for the initial core, the powder pellet route will be used for the fabrication of $(Th\text{-}Pu)O_2$ pellets due to the experience available. The $(Th\text{-}^{233}U)O_2$ pellets will have a central hole and will be fabricated by a pellet impregnation technique. In this process, a major part of the work will be carried out in low active areas.

8. DESALINATION

Utilisation of waste heat from AHWR for seawater desalination is a feature incorporated in the design. Integration of Low Temperature Evaporation (LTE) desalination plant with AHWR leads to the utilisation of a significant part of waste heat for producing the entire makeup demineralised (DM) water from seawater in an economical and environmental friendly manner. An LTE desalination plant utilises waste heat at as low as 323 K to convert saline water into pure water, rather than the chemical pretreatment of raw water and ion exchange process as in a conventional DM plant.

In the steam and feed system of AHWR, 124 kg/s of steam is taken out from the turbine to heat the feed water in HP and LP preheaters. After heating, the condensed steam at 344 K is cooled before passing into the main condenser. Around 15 MW_{th} heat is transferred to process water which goes as waste heat. To utilise the waste heat from condensed steam, heat is transferred from condensed steam to process water in an intermediate feed water-process water heat exchanger. Process water is heated up to 338 K. The part of heat (8 MWth) carried by process water from 338 K to 322 K will be utilised in desalination plant to produce 400m^3/day product water. Intermediate heat exchanger is incorporated between feed water and desalination plant to ensure that no radioactive material reaches desalted water. Process water is further cooled from 322 K to 308 K in process water-seawater heat exchanger.

9. FEATURES OF AHWR RELEVANT FOR ITS ECONOMIC COMPETITIVENESS

The reactor incorporates features to achieve enhanced economic performance and improvement in overall plant thermal efficiency through better management of energy produced in the core, elimination of expensive equipment, introduction of passive components reducing the need for repair and maintenance, and optimum fabrication route. Some of the features are listed below:

a) Use of excess heat generated in moderator and vault water to heat the feed water

b) Elimination of primary coolant pumps and associated prime movers

c) Replacement of steam generators with steam drums of simple construction.

d) Introduction of light water as coolant instead of heavy water

e) Substitution of several safety grade equipment with conventional equipment backing up a passive system.

f) Maximisation of shop fabrication to save reactor construction time

g) Introduction of easily replaceable coolant channels for ease of repair and maintenance.

CONCLUDING REMARKS

The design of the Advanced Heavy Water Reactor has progressed from concept, through feasibility assessment and preliminary design stages, to detailed design, design optimisation and design validation stage [7-9]. Development activities in different areas have been undertaken. A critical facility and an integral test loop are being set up for the validation of physics and thermal hydraulic codes respectively. Tests are planned to confirm coolant channel replacement and refueling technologies. Existing expertise is being complemented with continued development efforts to successfully implement the programme related to fuel cycle. In parallel with the analytical and experimental work, work is on for the preparation of a Detailed Project Report (DPR) and Preliminary Safety Analysis Report (PSAR).

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