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International Conference on Opportunities and Challenges for Water Cooled Reactors in the 21st Century

Vienna, Austria, 27–30 October 2009

BOOK OF EXTENDED SYNOPSES

Painting by Jerrika C. Shi, Aged 10

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Organized by the

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EC Joint Research Centre (EC/JRC)

International Electrotechnical Commission (IEC)

OECD Nuclear Energy Agency (OECD/NEA)

World Nuclear Association (WNA)

PROGRAMME OVERVIEW

Time	Monday 26 Oct. 2009	Tuesday 27 Oct. 2009	Wednesday 28 Oct. 2009	Thursday 29 Oct. 2009	Friday 30 Oct. 2009
09:00 - 10:30		Registration *(08:00 – 09:30)* **Opening Session** “Current Nuclear Power Outlook: Opportunities & Challenges" *(09:30 – 11:10)*	**Plenary Session** “Design and Construction of Advanced Water Cooled Reactors” (Part II)	**Plenary Session** “Safety and Performance Excellence in Current NPPs”	**Panel Discussion** ”Role of Water Cooled Reactors in the 21st Century”
10:30 –			Coffee Break		
11:00 - 12:30		Coffee Break *(11:10 – 11:30)* **Plenary Session** “Challenges in Near Term Nuclear Power Deployment” *(11:30 – 13:00)*	Topic 1 **Sess. 1-2** Topic 2 **Sess. 2-2** Topic 3 **Sess. 3-2**	**Plenary Session** "Advanced Applications of Water Cooled Reactors"	Topic 2 **Sess. 2-3** Topic 4 **Sess. 4-3** Topic 5 **Sess. 5-3**
12:30 – 14:00		Lunch *(13:00 – 14:00)*	Lunch		**Closing Session**
14:00 - 15:30		**Plenary Session** “Design and Construction of Advanced Water Cooled Reactors” (Part I)	**Panel Discussion** “Advanced Monitoring and Diagnostic Technologies in NPPs”	**Poster Session & Technical Exhibitions**	
15:30 –		Coffee Break			
16:00 – 18:00	Regi-stration (16:00 – 19:00)	Topic 1 **Sess. 1-1** Topic 2 **Sess. 2-1** Topic 3 **Sess. 3-1**	Topic 4 **Sess. 4-1** Topic 5 **Sess. 5-1** Topic 6 **Sess. 6**	Topic 4 **Sess. 4-2** Topic 5 **Sess. 5-2** Topic 7 **Sess. 7**	
18:30 –		Welcome Reception (Place : M building)		Official Dinner (Place : Melker Stiftskeller Vienna)	

Topic 1. Current Nuclear Power Outlook– Opportunities to Launch New NPP Programme
Topic 2. Nuclear Deployment Challenges and Solutions–Institutional and Cross Cutting
Topic 3. Design and Construction of Advanced Water Cooled Reactors
Topic 4. Safety and Performance Achievement in Current NPPs
Topic 5. Advanced Technology Application
Topic 6. Safety Assessment in NPPs
Topic 7. Instrumentation & Control

LIST OF EXTENDED SYNOPSES

Part 2 Synopses for Oral Presentations

TOPIC 3: DESIGN AND CONSTRUCTION OF ADVANCED WATER COOLED REACTORS

TOPIC 4: SAFETY AND PERFORMANCE ACHIEVEMENT IN CURRENT NPPS

TOPIC 7 : INSTRUMENTATION & CONTROL

Part 3 Synopses for Poster Presentations

TOPIC 1: CURRENT NUCLEAR POWER OUTLOOK: OPPORTUNITIES TO LAUNCH NEW NPP PROGRAMME

TOPIC 2: NUCLEAR DEPLOYMENT CHALLENGES AND SOLUTIONS - INSTITUTIONAL AND CROSS CUTTING

TOPIC 3: DESIGN AND CONSTRUCTION OF ADVANCED WATER COOLED REACTORS

TOPIC 4: SAFETY AND PERFORMANCE ACHIEVEMENT IN CURRENT NPPS

Part 1

Synopses for Keynote Presentations

IAEA's Support of Water Cooled Reactors in the 21st Century and Beyond

K.S. Kang, S. Bilbao y León, O. Glockler

International Atomic Energy Agency (IAEA), Austria

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In recent times, there has been a two prong approach on the expansion of nuclear power. On the one hand, countries with existing nuclear power programmes have made a large effort towards making the most of their current nuclear assets by capitalizing in many years of operational excellence, as well as by extending and optimizing their operational life. On the other hand, and despite these Plant Life Management efforts, there is a clear need to eventually replace current nuclear capacity and also to meet increased energy demand in an environmentally sound manner. These interests have motivated both countries with existing nuclear programmes and newcomer countries to consider the construction of new nuclear power plants (NPP) to fulfil those needs. In the near term, these new NPPs will most likely be advanced Water Cooled Reactors (WCR) that are the result of combining current operational experience with additional innovations resulting from worldwide research and development.

When it comes to Plant Life Management (PLiM), the safety considerations of a NPP are paramount and those requirements have to be met to obtain and to extend/renew the operating license. To achieve the goal of the long term safe, economic and reliable operation of the plant, a plant life management programme is essential. Some countries already have advanced PLiM programmes while others still have none. The PLiM objective is to identify all that factors and requirements for the overall plant life cycle. The optimization of these requirements would allow for the minimum period of the investment return and maximum of the revenue from the sell of the produced electricity.

Recognizing the importance of this issue and in response to the requests of the Member States, the IAEA Division of Nuclear Power implements the Sub-programme on "Engineering and Management Support for Competitive Nuclear Power". Three projects within this sub-programme deal with different aspects of the NPP life cycle management with the aim to increase the capabilities of interested Member States in implementing and maintenance of the competitive and sustainable nuclear power.

Although all three projects contain certain issues of PLiM, there is one specific project on guidance on engineering and management practices for optimization of NPP service life. This particular project deals with different specific issues of PLiM including aspects of ageing phenomena and their monitoring, issues of control and instrumentation, maintenance and operation issues, economic evaluation of PLiM including guidance on its earlier shut down and decommissioning.

As the nuclear community worldwide looks into the future with the development of advanced and innovative reactor designs and fuel cycles, it becomes important to explore the role Water Cooled Reactors will play in this future. To support the future role of WCRs, substantial design and development programmes are underway in a number of Member States to incorporate additional technology improvements into advanced NPP designs. In synergy with countries efforts in technology development in support for near-term deployment of new NPPs, the IAEA Division of Nuclear Power implements the Sub-programme “Technology Development for Advanced Reactor Lines”. Two projects within this sub-programme focus on fostering the collaboration and the exchange of information among Member States towards the technology advance of WCRs for improvements in performance, economics and safety

One of the key features of advanced nuclear reactor designs is their very high level of safety due to a reduction in the probability and consequences of accidents and to an increase in the operator time allowed to better assess and properly react to abnormal events. A systematic approach and the experience of many years of successful operation have allowed designers to focus their design efforts and develop safer, more efficient and more reliable designs which meet very stringent safety standards, and to optimize plant availability and cost through improved maintenance programs and simpler operation and inspection practices.

The paper describes some of its achievements in IAEA activities on different issues of PLiM and advanced WCR during the nearest past as well as plans for the future.

European Nuclear Safety Research for the Nuclear Renaissance

Giovanni De Santi

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Joint Research Centre, European Commission

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The paper explores the potential for advanced research programmes as support to the development of energy related scenarios in the near term. In particular, being the energy sector identified as a major contributor to climate change, research is called to play a major role in combining safe technologies with sustainable solutions. Nuclear technology will represent a viable solution to meet energy demand, security and sustainability targets. Still for a long time, research will be called to ensure safe and reliable operation of nuclear installations, both current and future, for an extensive period of time and regardless of geographical distribution.

The European Commission has a long history in nuclear research, since the Euratom Treaty. Key developments in European nuclear fission research in recent years include:

- The Sustainable Nuclear Energy Technology Platform (SNETP), comprising strategic research, deployment, training and funding strategies.
- Euratom participation in the Generation IV International Forum and as an observer to the US Global Nuclear Energy Partnership (GNEP).

European research currently focuses on:

- Improving safety and reliability aspects of current nuclear installations.
- Developing safe and efficient design for future generation power plants.
- Providing solutions for nuclear waste minimization and disposal.
- Easing operation, in particular enhancing maintenance.
- Predicting plant lifetime and extending the available safe operating life.
- Maintaining critical nuclear competences.

The paper will explore current and future trends in research supporting nuclear energy safety and security, providing preliminary evidence on the validity of the proposed approach and case studies on selected topics.

Dr Giovanni De Santi graduated at the University of Pisa (Italy) and obtained his PhD at the Fluidodynamics Technical University of Munich (TUM). Since 1985 he has been employed at the Joint Research Centre (JRC) of the European Commission, where he developed his professional expertise in the fields of energy, mechanical/chemical engineering and environmental science. In 2007 he was nominated Director of the Institute for Energy of the JRC based in Petten (The Netherlands) and Ispra (Italy). Dr De Santi is member of the Steering Committee and/or Governing Board of numerous European Commission Programmes, such as the European Strategic Energy Technology Plan (SET-Plan), the Hydrogen and Fuel Cell Joint Technology Initiatives, European Technology Platforms (Solar Energy, Biofuels, etc.) as well as Chairman of international Task Forces such as the UNECE-OECD Task Force for heavy duty transport emissions, etc.
He is the author of many peer reviewed publications in high level scientific journals and in international conferences.

The Challenge and Countermeasures for Human Resources Development on Nuclear Power in 21st Century

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Most IAEA member states have set up nuclear power development program as national policy and energy strategy with the technical progress and successful demonstration of nuclear power to be a safe clean, ecological and economic energy. The development and supply of Human Resources (HR) is becoming the big challenge in the effective and sustainable development of nuclear power as all countries in the world are in short of professional nuclear engineers and skilled technician at present. The problem is resulting from firstly many universities have closed or reduced nuclear power related faculty after two main severe nuclear power accidents, secondly the existing nuclear power engineers and technicians are going to retire and we have difficulties in finding replacement timely, thirdly the training period for nuclear power professionals is much longer than those of the other industries.

The HR bottleneck situation should be seriously considered for the safe and economic development of nuclear power. How to confront and find a practical solution to the challenge? I suggest member states firstly to formulate an appropriate HR training program before drawing up any nuclear power development plan; secondly to make nuclear power technology education network or set up nuclear power university in different stages and regions thus to provide distance learning and further study for the existing professional and skilled technician; thirdly to promote international cooperation for nuclear power peaceful utilization in a win-win strategy; fourthly to organize international nuclear expert forum and communication platform to maximize their talent and speed up the development in nuclear power manpower supply as well as establish a world-wide nuclear power HR foundation.

SNERDI/SNPTC of China has adopted various countermeasures to face the challenge of nuclear power manpower by establishing and implementing the nuclear power human resource program. The main points are as follows: to introduce advanced talents, to upgrade scientific training system and to enhance nuclear power manpower training and administration control, to establish efficient human performance evaluation system, to provide related courses in Master and PhD degree, to provide guidance for staff development and future advancement through nuclear power educational network and university, and etc .

Nuclear Power: an Irreplaceable Option for Sustainable Development

Philippe Pradel

Director
French Atomic Energy Commission

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Many signs of a global renewed interest in nuclear energy can currently be observed worldwide. A vision for the place of nuclear energy in the future is one of a prominent role throughout the 21st century, as this energy perfectly matches the requirements for the sustainable development of mankind. Nuclear energy has the ability to contribute significantly to world energy needs without producing greenhouse gas emissions.

Light Water Reactors (LWR) will play a leading role in the Nuclear Renaissance and the industry is preparing actively to face the great need for the supply of reactors. However, commitments and international obligations in terms of safety, security and non-proliferation standards will have to be strictly observed: the highest level of safety and the present international harmonization initiative shall be fostered, and the organization of back-end fuel cycles along with spent fuel and radioactive waste will have to correspond to a responsible, secure and non-proliferant form of management. Fuel services in particular, using the best available proven technologies, should be set up under the IAEA umbrella.

In order to increase the sustainability of nuclear energy, fourth generation systems will need to be developed. The french R&D strategy is based on the development of fast reactors with closed fuel cycles along two tracks: the Sodium Fast Reactor (SFR) and the Gas fast Reactor (GFR), with advanced processes for spent fuel treatment and recycling.

This challenge will be achieved thanks, in particular, to international collaboration and dialogue. As it has been expressed at several occasions by President Sarkozy himself, France finally intends to play a part in the coming nuclear renaissance, by assisting any country willing to develop its own nuclear capacities with its unanimously recognized experience, provided of course that it respects international non-proliferation and safety commitments.

***Philippe PRADEL**, 53 years old, a graduate of France's leading engineering school Ecole Polytechnique began his career with the French Atomic Energy Commission (CEA) as a research scientist on the SUPERPHENIX liquid metal fast breeder reactor and was part of the team that started up that reactor.*

Mr. PRADEL joined Cogema in 1987 and became in 2003, Senior Executive Vice President of COGEMA, in charge of Treatment, Recycling and Logistics. From 2005 up to 2009, Mr. PRADEL was the Director of the Nuclear Energy Division at the French Atomic Energy Commission (CEA), in charge of the whole nuclear energy sector.

Since 2006, Mr Pradel is member of the board of AREVA, and chairman of the European Sustainable Nuclear Energy Technology Platform since 2007.

Challenges Faced by Developing Countries in Nuclear Power Deployment

Hamad AlKaabi

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Permanent Representative of the United Arab Emirates to the International Atomic Energy Agency

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Analysis of future domestic electricity demand and supply conducted by official UAE entities has concluded that increasing demand for electricity is fast outstripping the growth in supply. Total electricity demand in the UAE is expected to rise from approximately 15,000 megawatts to 42,000 megawatts by 2020. Significant new generation capacity must be constructed and brought on-line. It was concluded that peaceful nuclear power-generation represents an environmentally promising and commercially competitive option which could make a significant contribution to the UAE's economy and future energy security. To make clear its intentions with regard to nuclear power, the Government of the UAE has prepared and formally endorsed its "Policy on the Evaluation and Potential Implementation o Peaceful Nuclear Energy" as a reflection of its views on the potential establishment of a peaceful civilian nuclear energy program. The policy defines the framework under which the program will be developed and is based on principles of transparency, highest standards of safety, security and non-proliferation , and working directly with the IAEA and responsible nations of expertise.

Many challenges face developing countries embarking on the development of a civil nuclear energy program. Challenges include initial questions such as where and when a nation should start planning. Other challenges are related to the development of required infrastructure in legislation, regulatory, human resources, and institutional structure. Further challenges are faced at the time of transforming guidance and recommendations into an implementation plan and the execution of such plan in an effective manner. The UAE has addressed many of these challenges by conventional and sometimes innovative ways in developing the required infrastructure and moving into the implementation phase of the program. Starting from almost no nuclear energy infrastructure, these plans are being conducted today in the UAE with an expanding number of domestic and international stakeholders.

AlKaabi has helped lead the energy assessment, and has served as a primary interlocutor on matters relating to nuclear energy and nonproliferation between the UAE Government and international organizations and other governments. He is the UAE's permanent representative to the International Atomic Energy Agency (IAEA) and was named in April 2008 by the UAE Foreign Minister as the Special Representative for International Nuclear Cooperation. He has been personally involved in all key milestones of the nuclear energy assessment, including a national scope energy assessment evaluating future UAE requirements and potential sources of electricity ,and drafting and release of the "Policy of the UAE on the Evaluation and Potential Development of Peaceful Nuclear Energy". Ambassador AlKaabi was trained as a nuclear engineer, getting his bachelor’s and master’s degrees in nuclear engineering from Purdue University in Indiana (USA). His graduate work focused on nuclear safety.

Advanced Nuclear Power Plants in Korea

Jong Shin Kim

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Korea Hydro & Nuclear Power Co., Ltd (KHNP) is the largest power company among the six subsidiaries that separated from Korea Electric Power Corporation (KEPCO) in 2001, accounting for approximately 25% of electricity producing facilities, hydro and nuclear combined.

KHNP operates 20 nuclear power plants in Kori, Yonggwang, Ulchin and Wolsong site and several hydroelectric power generation facilities, providing approximately 36% of the national power supply. As a major source of electricity generation in Korea, nuclear energy contributes greatly to the stability of national electricity supply and energy security.

KHNP's commercial nuclear power plant operation, which started with Kori Unit 1 in 1978, has achieved an average capacity factor more than 90% since 2000 and a high record of 93.4% in 2008.

Following the introduction of nuclear power plants in the 1970's, Korea accumulated its nuclear technology in the 1980's, developed OPR 1000(Optimized Power Reactor) and demonstrated advanced level of its nuclear technology capabilities in the 2000's by developing an advanced type reactor, APR 1400(Advanced Power Reactor) which is being constructed at Shin-Kori Unit 3&4 for the first time.

By 2022, KHNP will construct additional 12 nuclear power plants in order to ensure a stable power supply according to the Government Plan of Long-Term Electricity supply & Demand. 4 units of OPR 1000 reactor model will be commissioned by 2013 and 8 units of APR 1400 are under construction and planned. At the end of 2022, the nuclear capacity will reach 33% share of total generation capacity in Korea and account for 48% of national power generation.

Mr. Kim received a bachelor's degree in Mechanical Engineering from Seoul National University in 1972 and a master's degree in Business Administration from Ajou University in 2008. He began his career at Korea Electric Power Corporation (KEPCO) in 1972 and has held many management positions in the Korean nuclear industry e.g. Chief manager of KEPCO Paris Office in 1987, Vice President of Overseas Project Department and Power Generation Department in 1997 and 1998 respectively, Site Vice President of Kori Nuclear Power site in 1999 and Executive Vice President of Power Generation Division in 2001.

Mr. Kim was appointed President and CEO of KHNP in 2007. His work at KHNP focuses on strengthening safety of nuclear power plants, improving operation performance and promoting self-reliance of nuclear technology such as OPR1000 & APR1400.

ABWR Technology & Construction Experiences

Masaharu Hanyu

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Advanced Boiling Water Reactor, called ABWR, was developed as a plant of Improvement and Standardization Program, Phase III, by the Ministry of International Trade and Industry, Japanese Government. The basic design has been completed in cooperation with Hitachi, Ltd. Toshiba, corp. and General Electric, Co. in 1985. The construction of Kashiwazaki Kariwa nuclear power plant unit No.6 of Tokyo Electric Power Co. in Niigata JAPAN started in 1991. Our ABWR NPPs are only the third generation NPPs that have operation experience in the world.

The ABWR has been developed to achieve the following goals:

(1) Enhanced safety and reliability

(2) Enhanced operability and maneuverability

(3) Less occupational radiation exposure

(4) Improved economy

In order to attain these objectives, various technologies have been introduced and applied to ABWR. Among these technologies, most prominent devices are Reactor Internal Pump, Fine Motion Control Rod, RCCV and Control System installed of Advanced Human-Machine System. Now, four ABWR Nuclear Power Plant(NPP)s are under operation and two ABWR NPPs are under construction.

Applying advanced technologies to ABWR continuously, it was realized to make the construction period shorter than current one by introducing Module process with big crawler crane.

Based on our construction and operation experience of the third generation ABWR, we continue to improve reliability and safety of ABWR and apply them to the future ABWR of high efficient energy generation, in much shorter construction period but for longer operation life.

Such enough experience to build and manage ABWR NPP makes us confident to contribute any supplying technology and know-how of ABWR to the world.

Mr. HANYU Masaharu joined Hitachi, Ltd. Hitachi Works in 1975. He has been promoted his job in the field of Nuclear Plant Service Engineering including ABWRs until 2004. He has contributed to Nuclear Systems as a General Manager for two years and he was appointed Corporate Officer of Hitachi, Ltd. In 2007, he was appointed President & Representative Director of Hitachi-GE Nuclear Energy, Ltd.

Experience of ABWR Operation and Global Deployment

Takeo Shimizu

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The ABWR (Advanced Boiling Water Reactor), which has received Design Certification from the U.S. NRC, is the latest BWR. It is the only third generation nuclear power plant now in operation, and has proved its safety, reliability and consistent operation. Kashiwazaki-Kariwa Nuclear Power Station units 6 and 7, operated by Tokyo Electric Power Co. in Niigata, Japan, were the first ABWRs that we constructed. They returned to full-scale operation in August 2009, following the Niigata-Chuetsu-Oki Earthquake. This quake provided unasked for verification of the design and construction reliability of the ABWR.

ABWR development began in 1978, with a basic design drawn up by an international consortium of BWR suppliers. A final, detailed design was completed by Toshiba, Hitachi and General Electric, based on evaluations of the initial design. The development project began as the Plant Improvement and Standardization Program Phase III and was supported by the Japanese government. Reliability tests of newly developed equipment were performed with support from the Japanese government and utilities, and the results indicated the reliability of the new equipment and its suitability for use. The ABWR adopts automated operation to assure better operability, inspection, and replacement of equipment, for the purpose of reducing radiation exposure.

Four ABWRs are in operation in Japan, two are under construction and eight are planned. The ABWR has evolved with the introduction of more advanced technologies in Digital Instrumentation and Control (I&C), the Turbine System and Safety Systems.

Outside Japan, two ABWRs are under construction in Taiwan. We are also implementing plans to deploy ABWR for Units 3 and 4 of the South Texas Project (STP) nuclear power plant in the U.S., based on the results of a TVA study supported by the U.S. DOE. We are also deploying the ABWR in conformity with European nuclear regulatory requirements, in particular Finnish safety regulation. We are confident that applying advanced technologies and meeting regulatory requirements in countries planning to introduce the ABWR will allow deployment on the global scale.

In March 1977, Mr. Shimizu received his master’s degree in Mechanical Sciences and Engineering at Tokyo Institute of Technology Graduate School of Science and Engineering.

In April 1977, he joined Nuclear Energy System & Services Division at Toshiba. Since that time he has served in design and development on BWR and ABWR plant nuclear systems.

From April 1990 to 1991 he had engaged in SBWR project at GE.

In 1999, he was appointed as a senior manager of LWR Plant Project Engineering Department of Toshiba and he supervised nuclear plant construction projects.

Since April 2003, he has been a Senior Fellow and supervises Toshiba's LWR technology.

Since 1977 he has experienced many plant's design and construction totally.

VVER Reactors : Clean and Reliable Source of Energy in the Past and in the Future

N.B. Trunov

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The paper covers the position of VVER technology and the role it plays in the development of the world nuclear power industry, touching upon the technology-related key features. The main technical and economic indices achieved are considered resulting from the analysis of a 45-year operating experience and the ways to improve them. The principal tendency traced is the higher requirements for NPP economic efficiency with safety assurance level preserved up to the standards of the current regulatory documents. However, the enhancement of the economic efficiency is connected with both the NPP and reactor performance, namely, the power Unit installed power, efficiency factor, capacity factor, breeding factor etc.

Another tendency in nuclear power industry development is meeting the needs of the Customers in constructing NPPs within the power line from 100 to 1600 MW considering the expansion of the export of nuclear power technologies. The paper also covers the design features of small-size (300 MW.el). and medium-size (400 – 600 MW.el) power reactors. Large-size reactor designs cover the reactors for NPPs with electric power from 700 to 1600 MW. The paper briefly describes design concepts and the stages of design elaboration and implementation. The modernization of the basic designs of V-392 reactor and NPP-92 Project that hold EUR certificates, is shown to have an evolutionary nature. Besides, prospects for the future and the avenues of further development are considered for large power reactors. VVER-600 design, that is a two-loop reactor based on AES-2006 reactor design with electric power 600 MW and a two-loop VVER-1200A reactor plant are presented in the paper.

The proven technology and the infrastructure of VVER allow offering the concept of a supercritical pressure water coolant reactor (VVER-SCP). In comparison with the traditional VVER reactors the design is expected to offer major advantages in technical and economic indices due to a high thermodynamic efficiency factor (up to 45 %) and increased breeding factor with an orientation towards a closed fuel cycle.

The conclusions made describe the significant potential for VVER technology development to solve the problems of power industry in the long term.

Nikolay Trunov graduated from Moscow Power Engineering Institute in 1982. Doctor of Sc. (Eng). Author of more than 100 publications, including book "Hydro-dynamic and thermal-chemical processes in steam generator of NPP with WWER". Since 2001 holds position of Chief designer- Head of division of OKB "Gidropress", Podolsk, Russia. Main activities:

design, theoretical and experimental research of steam generators, heat exchangers and other components of NPP of different types.

Advanced Design of Mitsubishi PWR Plant for Nuclear Renaissance

Etsuro Saji

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Mitsubishi Heavy Industries, Ltd (MHI), has been playing the leading role in PWR technology in Japan for about 50 years. We have constantly contributed to the design, manufacturing and construction of all of the 26 PWRs in operation, under construction or under licensing in Japan, in spite of the recent stagnation of the new nuclear construction over the world. We have enhanced our PWRs safety, reliability, operability and maintainability throughout our history, collaborating with our customers and the government, such as in the "Improvement and Standardization Project of Light Water Reactors". We polished up the outcome of this joint effort of the public and private sectors, and materialized as the APWR plant. The Japanese first APWRs are under licensing as the Japan Atomic Power Company's Tsuruga units 3 and 4.

In parallel with domestic activities, we began our globalization history as a supplier of nuclear components, such as Steam Generators (SGs), Reactor Vessels (RVs), Reactor Vessel Heads (VHs), Pressurizer (PRs), Reactor Coolant Pumps (RCPs) or Steam Turbines. Those experiences gave us the confidence in our technologies as well as competence as a global supplier.

The design of the Japanese APWR is based on the conventional 4-loop plant technologies, with which MHI has accumulated significant operating experience, scaled up to achieve higher electrical outputs in Japan. In addition to adopting these proven technologies, modifications are also made on the Japanese APWR to improve economy, safety, reliability, operability, and maintainability. For the advancement of the nuclear power technology in Japan, the Japanese APWR has been developed. Newly developed Japanese APWR technologies are fully tested, well-verified, and well-established. The Japanese APWR output of 1538 MWe is produced from the large capacities of the reactor core and other major components such as the steam generator, reactor coolant pump, and turbines.

The US-APWR and EU-APWR are fundamentally based on the established Japanese APWR plant with its latest technologies to improve plant efficiency such as the employment of large steam generators and turbines, and other minor modifications to meet U.S. utility requirements or European Utility Requirements (EUR). With an increased plant safety, reliability, and performance, construction costs are being reduced due to the benefit of economy of scale resulting from an increase in capacity.

For its counterpart for the US market, the US-APWR, the standard design certification (DC) application was submitted to the US Nuclear Regulatory Commission (NRC) in December 2007 and docketed in February 2008. It had the Nuclear Energy Institute categorize Mitsubishi as the 4th “Nuclear Plant Designer” in the US. The technical review process is in progress and the Final Safety Evaluation Report is expected to be issued in 2011. Last September, Luminant filed their combined construction and operating license (COL) application for Comanche Peak units 3 and 4, referencing US-APWR standard design.

We held the “EU-APWR Technical Seminar” for European utilities in Brussels. We are preparing the compliance assessment with European Utility Requirements (EUR) to deploy the APWR technologies in Europe. The EU-APWR is an evolutionary Generation-III+ design based on proven technologies of Japanese APWR. It realizes the world largest class output of 1,700MWe. The 14ft fuels create additional thermal margin to realize 24-month refueling cycle without deterioration of the fuel economy.

Mainly in order to serve the region where mid-sized plants are wished because of the condition about power demand and/or grid capacity, we are offering 1,100MWe class Generation-III+ PWR, i.e. ATMEA1 as well. The ATMEA1 is at the stage of its basic design performed by ATMEA, the joint venture of AREVA NP and MHI.

Our latest new build, Hokkaido Electric Power Co., Inc.’s Tomari unit 3 is now at the final stage of commissioning and its commercial operation is expected to begin this December. Tomari unit 3 preoccupied the evolutionaly technology of APWR such as full digital I & C, advanced control room, steam genarator with high corrosion resistrance and three dimensions design turbine blade. Tomari unit 3 realizes high reliability with full digital I&C system and advanced control room, large electric output by employing optimized turbine design with 54”long end blades and other advanced features. For total management over design to construction of Tomari 3, we employed the unified 3D CAD database. For containment vessel (CV) construction, we employed parallel technique of on-site upper head assembling and body construction then hoisted the whole upper head assembly onto the body using super heavy duty crane, which realized the CV construction period of as short as eight months.

Based on our long years experiences as the total PWR plant supplier in Japan and of supplying key components to the global market, we commit ourselves to contribute to “nuclear renaissance” by offering EU/US-APWR and ATMEA1 Gen-III+ PWRs, and Tomari 3 type PWR as a “total plant supplier to the global market”.

Application of Advanced Technology to Improve Plant Performance

H.M. hashemian

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Abstract

Advances in computer technologies, signal processing, analytical modeling, and the advent of wireless sensors have provided the nuclear industry with ample means to automate and optimize maintenance activities and improve safety, efficiency, and availability, while reducing costs and radiation exposure to maintenance personnel. A review of these developments and examples of their broad use in the nuclear power industry will be presented together with the financial and safety benefits that they have produced.

As the current generation of nuclear power plants have passed their mid-life, increased monitoring of their health is critical to their safe operation. This is especially true now that license renewal of nuclear power plants has accelerated, allowing some plants to operate up to 60 years or more. Furthermore, many utilities are maximizing their power output through uprating projects and retrofits. This puts additional demand and more stress on the plant equipment such as the instrumentation and control (I&C) systems and the reactor internal components making them more vulnerable to the effects of aging, degradation, and failure. In the meantime, the nuclear power industry is working to reduce generation costs by adopting condition-based maintenance strategies and automation of testing activities.

These developments have stimulated the development of on-line monitoring (OLM) technologies and new diagnostic and prognostic methods to anticipate, identify, and resolve equipment and process problems and ensure plant safety, efficiency, and immunity to accidents. This speech provides examples of these technologies and demonstrates how they can provide the nuclear industry with the means to meet regulatory requirements, comply with technical specification provisions, or resolve operational and maintenance issues.

H.M. Hashemian has a Doctor of Engineering in Electrical Engineering, and a Ph.D. in Nuclear Engineering. He has worked for AMS since 1977 when the company was founded. AMS specializes in process instrumentation, equipment condition monitoring, on-line diagnostics of anomalies in industrial equipment and processes, automated testing, and technical training. The company's headquarters are in the USA with representative in Austria, Spain, South Korea, and Switzerland.

Dr. Hashemian has written two books, six book chapters and over 200 technical papers. His books have been translated to Chinese, Japanese, and Russian.

Through AMS, Dr. Hashemian has worked for almost all the 104 nuclear power plants that are currently operating in the United States and many in Europe and Asia. Furthermore, he has presented seminars, training course, and lectures to nuclear industry professionals in 15 countries. He was elected a Fellow of the International Society of Automation in 1992 and is a member of the American Nuclear Society, Institute of Electrical and Electronics Engineers, and European Nuclear Society.

NULIFE – its Role in Implementing Strategic Research of LTO related to PLIM Issues in Europe

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The European network of excellence NULIFE (Nuclear plant life prediction) has been launched with a clear focus on integrating safety-oriented research on materials, structures and systems and exploiting the results of this integration through the production of harmonised lifetime assessment methods. NULIFE will help provide a better common understanding of the factors affecting the lifetime of nuclear power plants which, together with associated management methods, will help facilitate safe and economic long term operation of existing nuclear power plants. In addition, NULIFE will help in the development of design criteria for future generations of nuclear power plant.

NULIFE was kicked-off in October 2006 and will work over a 5-year period with over 40 partners drawn from leading research institutions, technical support organisations, power companies and manufacturers throughout Europe. NULIFE also involves many industrial organizations and, in addition to their R&D contributions, these take part in a dedicated End User Group. Four expert groups, with identified members and links to national programmes, have now produced state-of-the-art type reports related to their expertises. Stress corrosion cracking and thermal fatigue pilot projects have finished concluding reports. Several project proposals have been introduced and optimised for new NULIFE pilot projects or other R&D projects.

The importance of the long term operation of the plants has been recognized at European level, in the strategic research agenda (SRA) of Sustainable Nuclear Energy Technology Platform (SNETP). The SRA defines the strategic targets in long term operation (LTO), performance improvement and external factors. In the long term operation area, safety justification, ageing mechanisms of systems-structures-components, ageing monitoring and prevention and mitigation of ageing are important subjects. In addition, some generic cross-cutting focus areas like structural materials, prenormative research, codes and standards, modelling, simulation and methods are considered. In NULIFE, the near future action will be the preparation of road maps and specific short, medium and long term research topics for each strategic focus areas identified in the SRA.

Based on NULIFE business plan, the discussion of long-term business plan, operational model and statutes of the future NULIFE Association has been started. In this context NULIFE maintains the sustainability of nuclear power by focusing on the continued, 60+ years of safe operation of nuclear power plants. The recognition of NULIFE's position as a key instrument in EU wide strategy implementation will assist the establishment of the

NULIFE Association and providing the sustainable LTO and PLIM related research and harmonised procedures. This presentation will outline the operating model and structure as well as key highlights of the current and future activities of the NULIFE.

Hydrogen Production using Water Cooled Reactors

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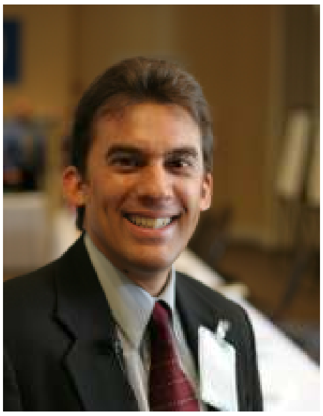
Today the world is facing tremendous energy challenges. There is a demographic explosion, which even in the most conservative scenarii will drive the energy demand to high levels whilst at the same time fossil resources are becoming scarcer, and more particularly oil which bears most of the weight in the transportation area. Global warming is also becoming a major concern as the last Intergovernmental Panel on Climate Change concluded that anthropogenic greenhouse gases (GHG) are responsible for most of the observed temperature increase since the middle of the twentieth century. To address these difficulties, the first step is to look for ways to save energy whenever possible. Then, the part of GHG free sources - renewable energies (wind, solar, hydraulic, biomass,..) as well as nuclear energy - has to be increased in electricity production. Lastly, since the part of electricity in the final consumption of energy is less than 20% worldwide, GHG free sources of energy have to look for other markets such as transportation, whether directly (electric cars) or indirectly via hydrogen (fuel cells,…) and/or process heat.

Hydrogen is produced currently from fossil fuels (less than 5% is produced by splitting water), and production is increasing steadily, mostly because of its use for refining crude oil and the more demanding standards of purity required. This alone is already stimulating interest in producing hydrogen by sustainable means. Moreover, the hydrogen market is bound to expand soon: hydrogen has been identified as a leading candidate for transport applications. A near term solution is to use the hydrogen produced together with a carbon source (biomass, coal, waste, CO_2, …) to make synthetic fuel. A longer term and more hypothetical development could be the direct use of hydrogen to power cars. Hydrogen could also be used in the iron and cement industries as a reducing agent and also help these CO_2 intensive industries to significantly decrease their GHG emissions.

The French context has also to be taken into account. More than 80% of electricity is produced by nuclear power plants and the hour to hour variations of the electricity demand have to be absorbed by having some plants operating at intermediate power over significant periods of time. This situation presents the double drawback of not taking full benefit of an expensive investment as well as having to take careful steps when going back to full power in order to preserve the fuel cladding. Hydrogen production during off peak periods could help regulate the electricity demand and operate the nuclear plants in base load. This then requires hydrogen production means that are flexible and not investment intensive, as they would be used only on a part time basis.

CEA's strategy is hence to focus on processes which could be coupled to nuclear plants or renewable energy sources and thus be able to produce hydrogen in a sustainable way, by splitting the water molecule using GHG free electricity and/or heat. Low temperature electrolysis, even if it is used currently for limited amounts, is a mature technology which uses only electricity and can be generalized in the near future. However, this technology, which requires about 4 kWh of electricity per Nm^3 of hydrogen produced, is energy intensive and therefore three advanced processes have also been investigated: High temperature steam electrolysis (HTSE), the Sulfur-Iodine (S-I) and Hybrid sulfur (HyS) thermochemical cycles. These processes look promising but the last two require the development of high temperature reactors, still necessitate extensive R&D work and will not be mature for industrial development within the next 20 years. Therefore, beside the optimization of LTE, our focus will be on HTSE, which will be available sooner and can also operate in autothermal mode, offering the capacity to be coupled to a LWR. In this paper, we will present the French roadmap for hydrogen production.



Innovative Water Cooled Reactor Concepts – Small and Medium Reactors

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Innovative water-cooled Small and Medium Reactors (SMRs) will play an essential role in meeting global energy needs. Based on well-known and well-tested water cooled reactor technology, this type of SMR can be licensed within the existing regulatory framework and have a ready supply of fuel and component manufacturers. As a result, water-cooled SMRs represent a near-term nuclear power solution. Excellent overviews of numerous SMRs, including water cooled concepts such as the KLT-40S, CAREM, SMART, IRIS, SCOR, AHWR, MARS, and MASLWR have been provided by the IAEA1,2. The advantages of SMRs are well known. Their smaller sizes make them suitable for a greater number of power grids and applications. This fact, coupled to their lower capital costs, yield a state-of-the-art nuclear energy solution with the potential of achieving a much greater market penetration than larger plants. SMR designs that utilize factory built components that can be shipped to a site by rail, truck or barge offer shorter construction times and greater cost certainty hence reducing financial risk. Many SMRs eliminate the choke points for the supply of large components such as reactor vessel forgings and large turbines. Their reduced core inventories present a smaller fission product source term which permits a greatly reduced emergency planning zone. This would allow SMRs to be sited in the proximity of large population centers while reducing power line losses. Some SMRs propose longer core lives which significantly reduce plant down time.

Among the water cooled SMRs, there exists a special class of reactors that are both modular and scalable. The fundamental concept is that of constructing a central power station comprised of multiple small power reactors. As regional power demand increases, additional modules can be added in the most economic manner to meet any size power need. Of particular interest is the NuScale Multi-Module Power Plant; a design that has evolved from the MASLWR concept developed at Oregon State University. This design employs both the economy of scale through the use of multiple modules and the economy of small through reduced size and simplification to offer significant operational advantages while being financially competitive with the large reactors. The NuScale plant design has been guided by a comprehensive set of customer utility requirements and informed by U.S. expert panels on reactor safety and data and analyses using state-of-the-art computer codes benchmarked against a 1/3-scale full-pressure, full temperature integral system test facility at Oregon State University. The unique characteristics of a NuScale multi-module plant are the topic of discussion for this paper.

The NuScale Nuclear Steam Supply System (NSSS) is very compact. It is enclosed in a containment that is only 18.3 m (60 ft) long by approximately 4.6 m (15 feet) in diameter. It includes the reactor pressure vessel which is 13.7m (45 ft) long by 2.7 m (9 ft) in diameter which contains the nuclear core, a helical coil steam generator, and a pressurizer. The nuclear core consists of an array of roughly half-height LWR fuel assemblies and control rod clusters at standard enrichments. The helical coil steam generator consists of two independent sets of tube bundles with separate feedwater inlet and steam outlet lines. Feedwater is pumped into the tubes where it boils to generate superheated steam. A set of pressurizer heaters is located in the upper head of the vessel to provide pressure control. Primary flow through the nuclear core is driven by natural circulation. Each module is independently connected to a small steam turbine generator set capable of producing a net power of 45 MW(e). The base platform for a NuScale plant consists of 12 modules capable of producing a total power of 540 MW(e). That is, one to twelve modules can be added to the base plant platform without the need for additional on-site construction. A multi-module central power station has many advantages. It eliminates single-shaft risk such that the loss of a single turbine does not shutdown the entire plant. Each module is identical and factory fabricated enabling major economic improvements through mass production. Factory production also enables the use of high performance replaceable NSSS components.

Each module resides completely underwater in a three-sided bay that is open to a common, stainless-steel lined pool located in the reactor building. The use of a single reactor building that houses a common underground cooling pool for all of the modules yields a unique set of economic, operational, safety and security advantages not previously recognized for this class of SMR. It allows for rapid and sequential refueling of each module. This keeps the overall plant availability very high such that only 45MW(e) are taken off the grid at any given time. Its unique refueling and maintenance bay minimizes refueling time and the size of the refueling and maintenance crews. Of major significance to safety and cost is the fact that the small diameter containment vessel significantly increases passive decay heat and containment heat removal while minimizing metal mass. This results in one of the safest and the most compact design for any multi-module plant concept. The simplicity and compactness offered by the design significantly reduces construction and material costs making the NuScale multi-module plant very competitive with large plants on a $/kW(e) of installed capacity. The NuScale multi-module plant design has successfully completed its first year of pre-application meetings with the United States Nuclear Regulatory Commission and is in active discussions with US utilities.

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Research and Development of Supercritical-Pressure Water Cooled Reactors

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Supercritical water does not exhibit a change of phase. The water density decreases continuously with temperature. The heat is efficiently removed at the pseudo-critical temperature, which is approximately 385°C at 25MPa. The low density fluid above this temperature is often called "steam" and high density fluid below it is called "water." Supercritical-pressure water cooled reactor (SCWR) adopts the once-through coolant cycle with the "water" as the reactor inlet and the "steam" as the outlet. The "steam" is directly fed to the turbines.

The advantages are the compactness of the plant system due to the high specific enthalpy of supercritical fluid, the simplicity of the plant system without the recirculation system and dryers of BWRs and steam generators of PWRs and high thermal efficiency without the limit of the boiling temperature.

Pressure vessel type SCWRs have been developed at the University of Tokyo since 1989 and is studied Japan, Europe and other countries. The University of Tokyo's version is called Super LWR and Super FR. The European version is called HPLWR (high performance light water reactor). Pressure tube type SCWR is studied in Canada and other countries.

Roughly speaking, the reactor pressure vessel and the control rods of Super LWR and Super FR are similar to those of PWRs, the containment and safety system are to BWRs and balance of plant is to supercritical fossil-fired power plants (FPPs).

LWRs were developed 50 years ago. Their successful implementation was based in part on experiences with subcritical fossil-fuel fired power (FPP) technologies at that time. The number of supercritical FPPs worldwide is larger than that of nuclear power plants. Considering the evolutionary history of boilers and the abundant experiences with supercritical FPP technologies, the supercritical pressure light water cooled reactor is the natural evolution of LWRs.

Water cooled fast reactors require a tight fuel lattice. The once-through coolant cycle is compatible with the tight lattice core of water cooled fast reactors. The increase in the core pressure drop due to the tight lattice does not cause problems with pumping power and stability because of the low coolant flow rate of the once-through cycle and small difference of water densities between the "steam" and the "water" at the supercritical pressure.

The plant system of the Super FR is the same as that of the Super LWR, a thermal reactor. Fast reactors do not need a moderator. Their power density is inevitably higher than that of thermal reactors. High power density is an advantage in economy. The Super FR has higher power density than the Super LWR. The Super LWR is expected to show better economy than LWRs due to the compactness, simplicity of the plant systems and high thermal efficiency.

The guidelines of the Super LWR and Super FR concept development are the following.

1. Utilize supercritical FPP and LWR technologies as much as possible.

2. Minimize large scale developments of major components.

3. Pursue simplicity in design.

The maximum temperature of the major components such as turbines, the reactor pressure vessel, main steam piping, reactor coolant pumps and control rod drives has been kept within the experiences of supercritical FPPs and LWRs. The concept of Super LWR and Super FR are developed based on numerical simulation at the University of Tokyo. The studies cover fuel and core design, plant control, start-up, stability, plant heat balance, and safety analysis. An advantage in safety is that depressurization cools the core in the once-through coolant cycle reactor such as Super LWR and Super FR.

Thermal hydraulic experiments using 7 fuel rod bundles are carried out at Kyusyu University and JAEA. The measurement of critical flow at depressurization, condensation of supercritical steam and the critical heat flux near the critical pressure have been made at Kyusyu University. Austenitic stainless steels based on the experience of PNC1520 are tested as the cladding materials at JAEA. Thermal insulation material, a type of zirconia was developed at the University of Tokyo. Water chemistry and dissolution of corrosion products in supercritical water are studied at the University of Tokyo.

The second phase of HPLWR project started September 2006 with FZK as the coordinator with EU funding. Design and integration, core design, safety, materials and heat transfer are studied with 10 European partners. Funding for research and development of the SCWR was begun in 2007 in China. Shanghai Jiao Tong University is the lead organization of eight partners. R&D of the pressure-tube type SCWR is conducted in Canada. NSERC/NRCan/AECL-University program started in 2008. The Coordinated Research Program (CRP) on heat transfer of supercritical fluid is organized by the International Atomic Energy Agency (IAEA).

Present paper includes the results of “Research and Development of the Super Fast Reactor” entrusted to the University of Tokyo by the Ministry of Education, Culture, Sports, Science at Technology of Japan (MEXT).

Part 2

Synopses for Oral Presentations

Synopses for Topic 1

CURRENT NUCLEAR POWER OUTLOOK: OPPORTUNITIES TO LAUNCH NEW NPP PROGRAMME

Approaches to Assess Competitiveness of Small and Medium Sized Reactors

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There is continuing interest in Member States concerning the development and application of small and medium sized reactors (SMRs), i.e., reactors with the equivalent electric power of less than 700 MW [1]. Currently, developed SMRs are in most cases intended for markets different than those in which large nuclear power plants operate. Such markets have essentially different investment requirements, siting flexibility, grid connections and infrastructure restrictions. Therefore, economic factors affecting the competitiveness or competition of SMRs in such markets would also be different from those observed in established markets for electricity production. For example, investment capability may be limited, which would favour capacity addition in smaller increments; grids may be small or weak, which would favour capacities exactly matching the demand; infrastructure and human resource may be insufficient, which would favour less complex operation and maintenance requirements; and non-electrical energy products, such as potable water, may be in demand, which would favour reasonably close plant location to the customer.

In practice it is futile to compare a single SMR needs to a single larger capacity plant on an economy of scale basis because SMRs are suitable for those locations that might not be appropriate for larger plants. However, a series of SMRs could be considered comparable to fewer larger plants to achieve the same overall power station capacity. In this case, SMRs have a potential to be competitive by employing alternative design strategies, taking advantage of smaller reactor size, offering a less complex design and operation and maintenance, relying on deployment-in-series approaches, taking an advantage of the accelerated learning, multiple unit factors and shorter construction duration. Reflecting on developments in member states, the International Atomic Energy Agency (IAEA) is finalizing the preparation of a report highlighting the economics and investment related factors affecting SMR competitiveness, as well as the tools available to assess these factors [2].

Adequate comparative assessment of SMR competitiveness is a complex task that may require considering time dependent expenditure and generation profiles and interest rates to determine adequate values of levelized unit energy cost (LUEC) for deployment scenarios with several consequently built plants of different types and capacities. As private capital may play an important role in financing of the new plants, investment and revenue models need to be employed to determine time-dependent characteristics, such as investments and revenues. Uncertainty analysis needs to be incorporated to add a degree of fidelity to the overall assessment. A variety of methods and tools required to conduct comparative economic assessments would been described in this paper. Many of these tools already exist in member states or are available from international organizations.

In addition to this, the IAEA coordinated the development of an open model for the analysis of SMR economic and investment opportunities, which targets bringing together all currently available state-of-the-art models for generation costs, revenues, financial costs, and external factors and risks, while "keeping the door open" for any new approach or development once it becomes available. The open model is being developed for a specific task of comparing the deployments of SMRs versus larger reactors in liberalized energy markets. LUEC will be an important figure of merit in this model; however, provisions would be made to ensure that LEUC is calculated taking into account time-dependent expenditure and production profiles and changing interest rates. In addition models to calculate investment profiles and revenues will be included. Finally, an approach to take into account other factors potentially affecting the competitiveness of SMRs, such as energy supply security, proliferation-resistance, political posture, etc. will be developed.

In parallel with the development of such model, IAEA is conducting a series of case studies to indentify options for SMR competitive deployment and relative benefits and disadvantages of smaller and larger reactors in different deployment strategies and application conditions, see Fig. 1. It can be seen that, when compared to a deployment of a single large reactor of the same capacity, staggered build of SMRs may minimize negative cash flow (capital-at-risk) but also may delay full site power availability to the grid and lower the net present value of the project by shifting cash inflows onward.

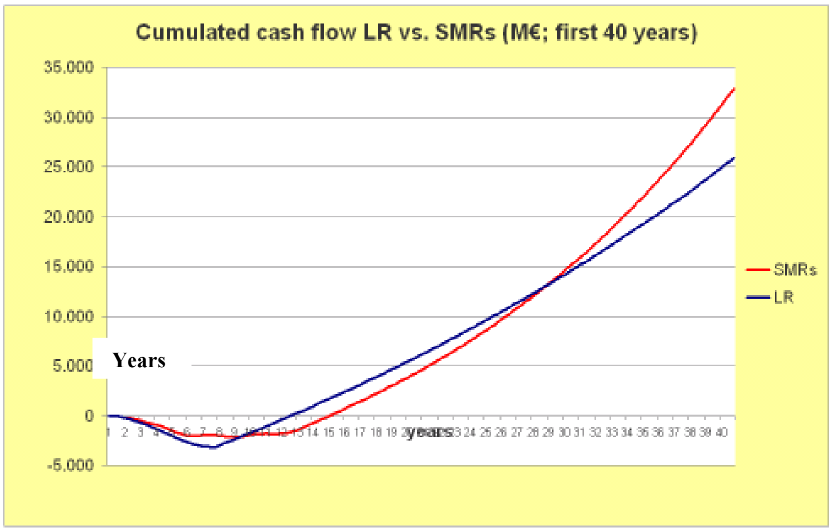

FIG. 1. Cumulative cash flow for a case study: staggered build of 4 SMRs versus one large reactor of the same overall capacity.

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Outlook of Nuclear Energy in Algeria

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The recent technological advances in the field of nuclear power plant design together with the continuous increase in the price of oil and gas and the search for lowering the global warming are making the option of using nuclear energy for the production of electricity and water desalination a very attractive alternative to be included in the energy mix of many developing countries.

The interests of Algeria in nuclear energy and its peaceful applications go back to the early seventies when a consistent program for training engineers and scientists in nuclear engineering was set up. Opportunity and feasibility studies for the construction of a nuclear power plant were conducted between 1975 and 1984 with the collaboration of IAEA. In parallel extensive efforts were made in the field of Uranium exploration and prospection.

In 1982 the 'Commission of New Energies' was created and undertook major actions that led to the implementation of basic nuclear infrastructures (nuclear research centers, research reactors, etc….).

The Chernobyl accident of 1986, the drastic drop in the price of petroleum and the economic recession that affected the country during the nineties strongly slowed down the progress of the Algerian nuclear power program.

The recovery of the economy and the strong increase in energy demand for the production of electricity and sea water desalination gave renew of interest in the nuclear option and prompted the creation of the 'Commission of Atomic Energy' in 1996 and its subsequent merging into the Ministry of Energy and Mines in the year 2006.

The past two years witnessed very noticeable progress in major fields aimed at energy planning, site selection and evaluation and the preparation of basic tools for the introduction by the year 2022 of a first nuclear power plant in the national energy grid.

Pertinent aspects of the Algerian nuclear power program are reviewed in this presentation.

Implementation of the Visaginas Nuclear Power Plant Project in Lithuania

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On June 28, 2007, the Parliament (Seimas) of the Republic of Lithuania passed the Law on the Nuclear Power Plant (No. X-1231) (hereinafter - the Law). This Law plays the role of the "decision in principle" for the construction of new nuclear power plant in Lithuania and the objective of this Law shall be to lay down provisions and to create legal, financial and organisational preconditions for the implementation of a new nuclear power plant project (hereinafter- Project). It is planned that the new nuclear power plant should start its operation in 2018. Implementing the provisions of the Law, the national investor, LEO LT, AB on 28 August 2009 has established the project development company "Visagino atominė elektrinė" UAB.

However, the preparatory phase of the implementation of the Project has commenced much earlier. Some of the preparatory activities are still ongoing but the finalization of the preparatory works is planned with the preparation of the procurement documents for the new NPP at the end of 2010.

The Pre-feasibility (Technology Assessment) study has been conducted in 2006 with the aim at identifying whether the construction of a new nuclear power plant in Lithuania is feasible from the legal, financial, technological and environmental and transmission grid point of view. The results of the Pre-feasibility study have shown that the implementation of the new nuclear power plant in Lithuania is feasible viewpoint of the aforementioned criteria.

Environmental impact assessment study has been performed and the corresponding report issued in the end of 2008. It successfully went through the public and international hearings and consultations. Environmental impact assessment procedure is going to be finalized in April 2009.

Transportation study activities are ongoing; the first part related to the investigation of the possible transportation routes for transportation of the heavy and oversized components to the construction site is already accomplished. The second part of the study, related to the economical and technical evaluation of the selected possible transportation routes has started.

The evaluation of the construction sites against the IAEA requirements has started in the beginning of 2009, during implementation of this project the evaluation of the two preselected sites against the IAEA requirements for seismic activity, geotechnical conditions, external human induced events and etc. is going to be performed.

The new power plant is going to be constructed near the Ignalina NPP which is currently running into its decommissioning phase, some of the infrastructure of the old plant can be used for construction of the new one. In order to correlate the decommissioning plans of the

Ignalina NPP with the construction of the new NPP a project for Ignalina NPP infrastructure overtake has been initiated.

In the frame of the Territory planning project the necessary arrangements in order to plan the land plots for new NPP construction are performed.

Drūkščiai lake hydrological and thermal balance measurements are performed in order to specify the data necessary for preparation of the project of the NPP cooling facilities and the evaluation of the lake thermal load.

Environmental audit of the construction sites is going to be performed in order to investigate the existing conditions of the sites and possible contamination stipulated by previous activities. The measures for site cleanup and resources necessary for this activity will be evaluated.

The project for the preparation of the infrastructure for connection of new NPP is aimed to prepare the technical conditions for the new power plant to be connected to the necessary engineering infrastructure (power lines, gas tubes, water lines and etc.).

With regard to development of legal environment, necessary changes in national legislation are being introduced with the aim to address the legal issues connected with development of Visaginas nuclear power plant project.

In order to meet the personnel requirements for a new NPP, the national specialist's education programmes in nuclear energy and nuclear energy physics were introduced in major universities of Lithuania.

The project is successfully proceeding forward with the major activities still lying ahead.

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On the Sustainability of LWR Development in China

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China's energy sustainability is facing three major challenges: (1) The defference between supply and demand; (2) The per capita energy resource deficiency; (3) The irrationality of energy mix.

Considered of the challenges, such as national security, energy safety and environment safety etc, taking into account the national oil and gas will be more dependent on international market; the coal-fired generation will put more and more pressure on the environment. The reasonable usable hydro power will be less and less. As a large-scale alternative energy source, nuclear energy has an excellent perspective in the Chinese energy development program because of its technology maturity, and its close to zero greenhouse gas emission. That's why nuclear power is the must in China.

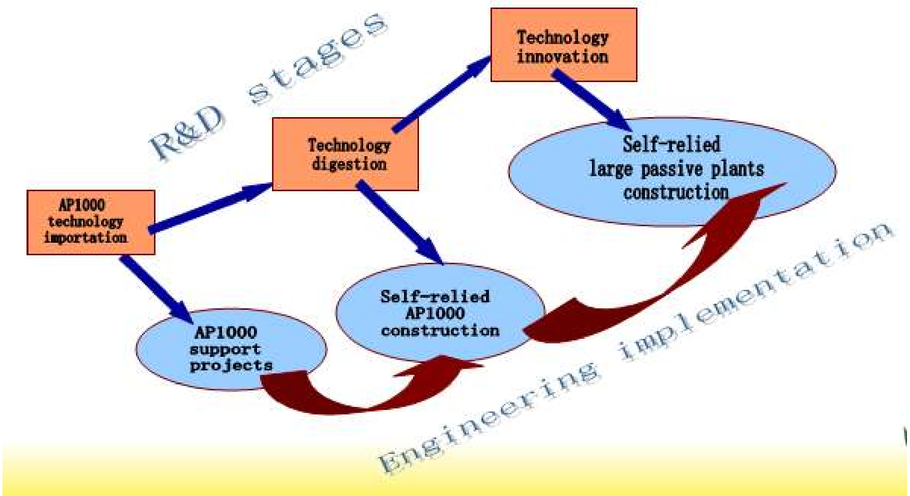
There is an old Chinese proverb-"opening the stream source, while cutting down the flow". China must take positive measures to open up new growth point in the energy supply, and gradually change current irrational energy mix. The reasonable pathway is to seek new increasing point of energy resource. Thus, nuclear power must be the most practicable option, and doubtless posses the irreplaceable strategic position. At the same time, it requires a comprehensive, coordinated and sustainable development of China's nuclear industry.

The 1st countermeasure to cope with nuclear power sustainability for meeting the urgent demand increase of nuclear energy is to import 3rd generation nuclear technology AP1000, speed up its self-reliance & innovation, improve its economical competitiveness, and commercially deployed as soon as possible.

At the beginning of this century, the government organizations initiated to make out the 2005-2020 National Sci. & Tech. Development Program, which was reviewed by Chinese Academies of Science, Engineering, and Society in 2002, and approved by State Council in 2003. Within the scope of this Program, a batch of National Key Projects was made up the mind to start over. Among them the project of "Large Advanced PWR and HTR Nuclear Power" is a superiority one. In terms of investment proportion its domain division is LAPWR, which has clarified to be carried out following the leading policy of "based on importation and digestion of world advanced nuclear power technology, to develop self-relied innovative PWR". It has pointed out that the China's larger scale development of nuclear power will start with a new great leap forward.

Authorized by the Government organization, SNPTC is the entity of the third generation nuclear power technology receiver, while acting as the chief bearer and the R&D platform to realize the G3 nuclear power technology self-reliance.

The 3 stage roadmap of LAPWR R&D is shown in this figure.



The world first AP1000 plant had its first concrete diffusion in April this year. It has declared a new ear of 3rd generation nuclear power development in China. The successive AP1000 nuclear power projects have been approved by government authorities. Further more, the inland NPP will start soon with the only option of AP1000, because of its higher safety goals and compact plant layout.

The self-relied larger passive plant finished its conceptual design last year end with the same safety goal as AP1000 but improved economical competitiveness. The relevant R&D activities have started recently.

The 2nd countermeasure to cope with nuclear power sustainability is to participate Generation 4 International Forum (GIF) for the future much larger scale development with focusing on main targets of sufficient use of nuclear resource, and nuclear waste minimization.

China has become the new signatory of GIF since 2007, and is now the member of VHTR System Steering Committee and the observers of both SFR and SCWR. Also, the Chinese experts are taking an active role in some IAEA's coordinate research projects related to SCWR. The Chinese government authority has approved the SCWR R&D Project within the frame of National 973 Program. 3 subjects, including material, thermal-hydraulics, and reactor physics, with 7 work packages are in progress.

China has already the available technology foundations of PWR, SFR and HTR, but compared with the advanced GIF technical goals there still exist large gaps. The purpose for China to join GIF activities is to follow up the world advanced technical level and make joint effort in some common interested R&D projects, so as not only to catch up with the world advanced level in proper time, but also to have the great stamina for future sustainable development of nuclear power in China.

Regional Approach to the Introduction of Nuclear Power Plants in Africa: Challenges and Opportunities

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In the Africa Region only South Africa operates Nuclear Power Plants for electricity generation. But since 2005 about sixteen other African countries have made political commitment to develop Nuclear Power Programme for electricity generation or for desalination of sea water. Similar interests have been shown by Member States in other Regions. In this regard, it is significant to consider the development and the status of radiation safety infrastructure in the region over the past one-and a- half decades, vis-à-vis the IAEA Model Project on "Strengthening Radiation Protection Infrastructure", which was introduced in 1994. The main objective of the Model Project was not only to recommend but to work together, "shoulder to shoulder" with Member States, to eliminate the shortcomings in their safety infrastructure and control of radiation sources. The Model Project has been adjudged very successful. In addition to the improved level of radiation and waste safety infrastructure in the region, Member States in the Region have even taken steps to consolidate the gains of the Model Project by appreciating the networking and the peer review mechanism provided under the Model Project and have gone ahead in 2008 to establish a regional association of regulatory bodies, which is similar to those in the other regions. This is the Forum of Nuclear Regulatory Bodies in Africa (FNRBA), the Charter of which was launched in Pretoria, South Africa in March 2009.

This paper seeks to leverage on the lessons and the experiences garnered from the Model Project on the "Strengthening Radiation Protection Infrastructure " to develop a paradigm or template for developing a Regional Model Project on the Development of Nuclear Safety Infrastructure in Member States of the Agency seeking nuclear power for the first time. Already, this is one of the seven thematic working areas the Forum wants to address in the next three years. The requirements for nuclear safety in terms of political and financial commitments are very different from those of radiation safety infrastructure. It is however noted that issues of nuclear safety and security have both national and international responsibilities which must be addressed without jeopardizing sovereignty of States. But then issues of poverty alleviation and therefore financing of Nuclear Power Plants will better addressed and facilitated on a well articulated and coordinated regional platform. This in turn will ensure transparency and compliance with international standards for radiation safety, nuclear safety and safeguards.

Future of the Nuclear Power Generation in Hungary

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Development perspective of the Hungarian economy and the same time its vulnerability is defined by import dependence of the energy supply. This had been recognised in the energy policy approved by the Hungarian Parliament in 2008. Security, stable, predictable price and environmental targets are set by the energy policy. Security of supply and the economical targets might be achieved by diversification of energy sources and power generation technologies, also diversification of import markets.

The electric power generation industry of Hungary is diversified considering the technologies. The gas, nuclear and coal are the main sources at growing share of renewable technologies. However the system needs further development since big part of existing capacities is obsolete. The development options are defined by the EU obligations, national interests and the trends in the industry. The industry is oriented to low risk incentives, preparing new project on gas and neglecting the adverse effects of increasing gas-dependence. Development of renewable capacities is clearly motivated by the state subsidies.

An intensive development of renewable capacities is limited by the economical constrains. After recent temporary drop of power consumption a moderate economical growth can be expected and predicted for long-term with annual increase of electric energy consumption between 0.5 to 1.5 percent. Assuming that the projects for increasing of energy saving and efficiency of end-use and the new renewable capacities will cover the net increase of demand, 3000-4000MW new power generation capacity will be needed for the replacement of shut-down old plants. 2000MW replacement capacity would be needed additionally, if the existing nuclear capacity would be shut-down when the originally licensed term will expire.

The role of nuclear power generation in Hungary can be assessed considering the mentioned above conditions and expectations. The Nuclear Power Plant Paks has an enormous role in the domestic power generation system, since it is the largest, cheapest and cleanest production capacity in the country. The nuclear power plant is the most important capacity for ensuring the security of supply. The plant strategy is to operate safe and as long as economically reasonable. Reactor power up-rate implemented recently provides 8% increase of plant output due to utilisation of modernised fuel assemblies and some minor modifications. The payback of power up-rate is less than four years. The licensed term of operation is 30years which expire between 2012 and 2017.

An extension of operational lifetime is feasible due to robustness of design and good condition of the plant. Justification of safety of additionally twenty years of operation is in progress. Solid regulatory system exists for the control and approval of licence renewal. Business assessment shows that the extension of operational lifetime is a reasonable decision. Other conditions as intermediate storage of spent fuel and final repository of radioactive waste are manageable. Extension of operational lifetime is widely supported by public.

Considering the forecast of energy demand and development perspectives of power generation industry in Hungarian, and also the targets of the energy policy, building of new nuclear capacity seem to be reasonable and feasible. The new nuclear capacity will really contribute to the sustainable development, decrease the vulnerability of economy due to import dependence, and motivate the overall development of engineering and constructing industry.

The preliminary analysis show that two units of 1000-1600MWe can be integrated into the system between 2020 and 2025. Paks site is available with necessary infrastructure. The experience of nuclear operators, knowledge of engineering and scientific support organisations, and the legal system exist for the preparation, construction and licensing of plant. In a 30 March 2009 vote, the Hungarian parliament has given overwhelming preliminary support to a government proposal to begin the detailed preparation for the construction of new nuclear generating capacity at the Paks plant. In the paper the long-term operation of Paks Nuclear Power Plant as well as the conditions and perspectives of a new built plant will be discussed.

Infrastructure Development and Challenges to Launch Nuclear Power Programme in Thailand:

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In June 2007, the cabinet passed a resolution for Thailand's Power Development Plan (PDP 2007). It was mentioned in the plan that Thailand will have 2 x 1,000 MWe nuclear power plants in 2020 and another 2 x 1,000 MWe in 2021. The PDP 2007 was revised in March 2009 and it was agreed to change the nuclear power generation to only 1 x 1,000 MWe in 2020 and 2021 respectively due to the large excess capacity at present. Many activities related to development of infrastructures in order to support electricity generation using nuclear power are being executed.

Milestones for nuclear power program implementation has been developed using the IAEA document "Milestones in the Development of a National Infrastructure for Nuclear Power" with some amendments to suit country situation. According to the schedule, a lot of activities related to infrastructure establishment, feasibility study, utility preparation and public education & participation are being performed. Within the year 2011, various issues such as legal and regulatory systems and international commitment, industrial and commercial infrastructure, technology transfer and human resource development, safety and environmental protection, public information and public acceptance, preparation of the nuclear power utility establishment, etc. must be solved out and undertaken to assure the cabinet to make final decision to go nuclear.

The action plan for the first three years (2008-2010) is shown in Table 1.

There are many challenges for Thailand embarking of the nuclear power programme. For example, it is essential to plan for the establishment of a regulatory body at the national level to support and regulate the nuclear power plant industry. Currently, the application for a license and the monitoring of a power plant are administered by the authorities of various agencies under different ministries; hence the process is very time-consuming and overlaps with one another. The approach that the regulatory body and the authorities to issue licenses relevant to the nuclear power plant operation are merged into one independent body is under consideration.

The human resources development (HRD) is also a key process and is time consuming. Future action to develop human resources for the Pre-project is shown in Figure 1.

Activities	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019	2020
1. Study and appraise treaties, international agreements, international laws, and international standards relating to a nuclear power plant in order to elicit their essential elements which are necessary to the drafting of the Thai Legislation on a nuclear power plant													
2. Draft comprehensive nuclear laws													
3. Expand existing NRB													
4. Start to develop human resources for NRB													
5. Sign the required treaties and conventional laws)													
6. License for construction permit													
7. License for commission permit													

Table 1: Draft Action Plan for the first three years (2008-2010)

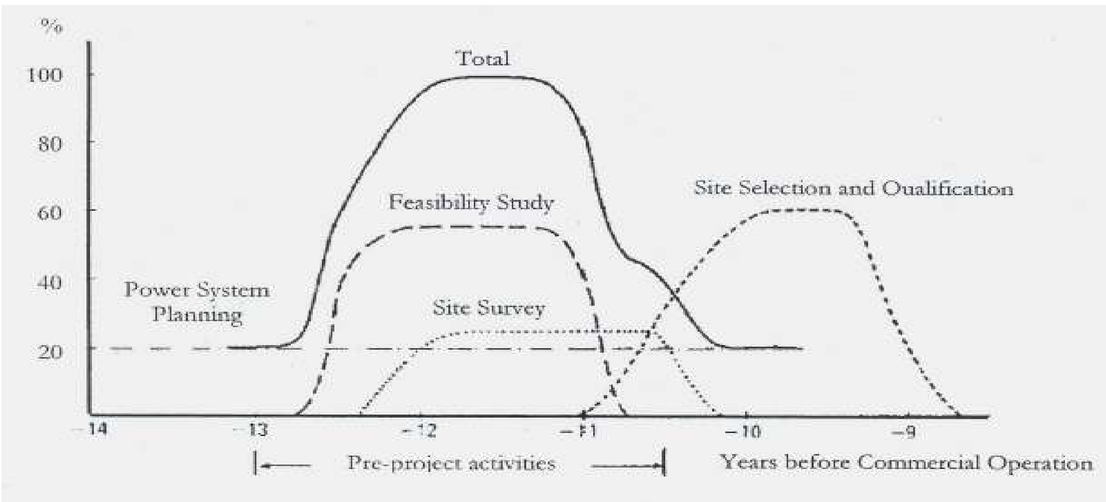

Figure 1: Proportion of HR Deployment for the Pre-project Activities

Evaluation on the Status of Indonesia Nuclear Infrastructure Development

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The demand for electricity increases every year. The increase is commensurate with the rate of increase in economic development and in population growth, and with the rapid developments in the industrial sector. To meet this demand for electricity, it is becoming more and more difficult to rely on existing resources which are limited. It is therefore very important that steps should be taken to seek other sources of energy supply as alternatives. Based on the premise that a Nuclear Power Plant (NPP) is technically safe, reliable, clean and environmentally-friendly, relatively economical, and supported by our modest achievements in preparations in respect of human resources and infrastructure, including the results of the feasibility studies for NPP development and the comprehensive assessment of different energy sources for electricity generation in Indonesia, the option of nuclear power could well be the right solution.

A nuclear power programme is major undertaking requiring careful planning, preparation and investment in term of time and human resources. As with any major programme, the commitment of resources to a nuclear power programme needs to be phased and decision to move to subsequent phases, where the commitment of resources will increase significantly, need to be made with a full understanding of the requirements, risk and benefits [1]. The milestone to identified three distinct phases in the introduction of a nuclear power programme and identified separate conditions for each : phase 1 covers the preparatory work in order to make an informed decision about a potential nuclear programme; phase 2 covers the development of the infrastructure issues required to be ready to begin and supervise construction of a nuclear plant; phase 3 covers the construction of the plant up to the approval to commission and operate [2].

In relation to the IAEA doc No. NG-G-3.1 year 2007, Indonesia has included nuclear power option within its energy policy as indicated by Presidential Decree No. 5 year 2006. Indonesia is ready to make commitment to a nuclear power programme by the issue of Act No. 17 year 2007, which states that the first nuclear power plant should be started in the year of 2015-2019 with high consideration of safety factor. In order to find out the readiness of Indonesia in the deployment of the first NPP, the status of its nuclear infrastructure development should be identified, and the result of the evaluation on the status of Indonesia nuclear power infrastructure development explain in the Table 1.

The aim of the evaluation approach is to : evaluate all relevant infrastructure issues in a consistent manner; bring the results together in order to identify a comprehensive action plan for moving into a subsequent phase of the establishment of a nuclear power infrastructure; provide a consistent international approach and enhance national competence through participation in a detailed and comprehensive evaluation. The 19 (nineteenth) of nuclear infrastructure are national position, nuclear safety, management, funding & financing, legislative framework, safeguards, regulatory framework, radiation protection, electrical grid, human resources, stakeholder involment, site & supporting facilities, environmental protection, emergency planning, security, nuclear fuel cycle, radioactive waste, industrial involment and procurement [3].

Table 1. The Status of the Indonesia Nuclear Power Infrastructure Development

No.	**INFRASTRUCTURE ISSUES**	**STATUS**
1.	National position	Minor Actions Needed
2.	Nuclear safety	Minor Actions Needed
3.	Management	Minor Actions Needed
4.	Funding and Financing	Minor Actions Needed
5.	Legislative Framework	Minor Actions Needed
6.	Safeguards	Minor Actions Needed
7.	Regulatory Framework	Minor Actions Needed
8.	Radiation protection	Minor Actions Needed
9.	Electrical Grid	No Actions Needed
10.	Human resources	Minor Actions Needed
11.	Stakeholder involvement	Minor Actions Needed
12.	Site and supporting facilities	Minor Actions Needed
13.	Environmental protection	Minor Actions Needed
14.	Emergency planning	Minor Actions Needed
15.	Security	Minor Actions Needed
16.	Nuclear fuel cycle	Minor Actions Needed
17.	Radioactive waste	Minor Actions Needed
18.	Industrial Involvement	Minor Actions Needed
19.	Procurement	Significant Actions Needed

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Synopses for Topic 2

NUCLEAR DEPLOYMENT CHALLENGES AND SOLUTIONS – INSTITUTIONAL AND CROSS CUTTING

Challenges and Opportunities to Introduce a First Nuclear Power Plant in Bangladesh

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Bangladesh is a country of 140 million people where 40% of people are access to electricity. Per capita energy consumption indicates the development growth of a country. Present energy consumption rate (165 kWh) of Bangladesh is the lowest among the developing countries. Over the years, due to the lack of indigenous energy resources, Bangladesh has become a gas dependent mono-energy based country. The indigenous gas reserve would be exhausted within 2017-2018 if it is used at the present rate for only power generation. There is no proven gas reserve due to lack of authentic survey. Eighty five percent of the total power plants are based on indigenous gas supply. It is therefore vital for the country to look for alternative fuel or power production and nuclear power technology is identified as a viable option in overall energy mix.

Due to limitation of natural resources and high demand for power for industrialization, poverty alleviation, millennium development goals (MDGs), Bangladesh Atomic Energy Commission (BAEC) has adopted a site on 292 acres of land way back in early 1960's at Rooppur in the Pabna district about 180 kilometers from Dhaka for implementation of the first nuclear power plant in the country. Generally, building a first nuclear plant takes longer time 8-15 years from inception in developing countries. To materialize the dream, a number of feasibility studies conducted so far has identified technical and economic merit of nuclear power in Bangladesh and that the site in general met the conditions for building a nuclear power plant. For introduction of a nuclear power plant, the stability of the national electricity grid is an important prerequisite. Necessary measures to minimize any fluctuation of frequency and voltage during normal operation have to be identified and implemented.

The Government approved the Bangladesh nuclear power action plan on 18th January 2000. As requested by the Government, the IAEA had sent several missions in several times, which had identified certain actions that are needed in the pre-implementation phase of building a 600-1000 MW nuclear power plant at Rooppur. The option is left open for building additional units in the future. The status of the actions is:

a. Updating Site Report /Feasibility study
   The site report has been updated and is being converted into a site safety analysis report following the review by the IAEA in 2003.
b. Nuclear Power Action Plan
   The Government has adapted a nuclear power action plan, which describes the principles and actions needed to facilitate implementation of the project. The plan was examined by the IAEA before the Government formally accepted it.
c. Request for Proposal

A draft request for proposal has been prepared, which will be used for inviting bids from intending suppliers and financiers.

It was not possible to make progress the project as expected due to various impediments, including financing as the most serious obstacle although the ECNEC had approved the project in 1980. Construction cost of a nuclear power plant (NPP) is 1.5-2.0 million USD per MW(e) whereas for a 600 MW(e) and 1000 MW(e) nuclear power plant, it costs about 1.5-3 billion and 3-4.5 billion USD, respectively. It is confirmed that generation cost of nuclear compared to fossil fuels is the lowest. It can be mentioned here that for a 1000 MW(e) power plant, 27 ton, 2.7 million ton and 2 million ton of nuclear, coal and oil fuels, respectively are needed in each year. The priority of the project is evident from the fact that there exists a Cabinet Committee, headed by the Head of the Government on the implementation of the project.

On the other hand, BAEC has been operating the 3MW TRIGA Mark-II nuclear research reactor for RI production, various R&D activities and man power training program since 1986. Regulatory body is the solely responsible for ensuring the safe use of radiation sources / equipment and the management of radioactive waste both in the public and private sectors under NSRC Act, 1993 and NSRC Rules-1997. The nuclear safety and radiation control act 1993 was passed in the National Assembly in 1993. The nuclear safety and radiation control division (NSRCD) is working with the IAEA for reforming, updating nuclear safety and radiation control rules for implementation of the Rooppur nuclear power plant in the country.

Moreover, Bangladesh has signed all treaties, convention, bilateral agreements, safeguards agreements, protocol additional to the safeguards agreements for peaceful applications of nuclear energy in the country. BAEC has constructed a central waste processing facility in the largest establishment of Atomic Energy Research Establishment (AERE), Savar, Dhaka for waste management arises from the research reactor and other nuclear medical centers located at different parts of the country.

A policy is also adopted to get a continuous supply of fuels and their front-end and back-end cycles in order to implement the first Rooppur nuclear power program in the country. The size and type of the reactor and grid capacity is in the process of negotiation with the vendors. Pressurised Water Reactor (PWR) or VVER type would be the potential candidate for selection due to its inherent safety features. Size should be justified after load flow and stability study. There is a plan to develop human resources in the field of design, construction and safety for successful implementation of the Rooppur nuclear power project. Several workshops, seminars, symposium and physical measures are being conducted by the BAEC for nuclear knowledge transfer, preservation and management against workforce ageing. Recently BAEC has introduced the safeguards, Safety and Security Division to oversee the country's nuclear materials accounting, control, safety and security systems.

Public acceptance is one of the important aspects for implementing the nuclear power program in the country. Due to the public acceptance and demand, the successful applications of nuclear technology for mankind in different sectors are substantially increasing day by day. Public acceptance of nuclear power program in the country is in general very good. Seminar, exhibition, press meeting/release, publication, public visit have been made on public information in order to achieve the public understanding and public acceptance on peaceful applications of nuclear energy.

Fuel availability, initial high investment, waste disposal, safety, long construction time and high decommissioning cost are the major concern but if overall per unit cost is competitive with coal and safety is ensured nuclear will be a prospective candidate for future power generation. Still there are many challenges; these include lack of adequate trained manpower for licensing of sites for nuclear power plants, licensing of design and construction of the plant and associated civil works and infrastructure and finally licensing the commissioning, operation and decommissioning of such nuclear plant facilities, it is equally important that our regulators must possess enough technological expertise to demonstrate competence in the strict adherence to the international safeguards regime. If Bangladesh is to get rid of chronic power shortage problem and look for long-term energy security and sustainable development, entry into a long-term nuclear power program should not be delayed anymore.

Nuclear Power Development, Financing and Delivery

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This presentation outlines the challenges facing the development of new nuclear power projects in particular in the emerging electricity markets.

The decision to employ nuclear power has always been a strategic government decision tied to national energy and infrastructure policy. The first generations of nuclear power plants were financed through government debt or sovereign guarantees. The government guarantees attracted favorable financing and repayment terms. In addition, the associated risks of construction and delivery were primarily secured by government backstops. These guarantee programs were necessary to support the size and risk of these projects that were integrated into government regulated energy markets. In some markets such as the US, the public utility boards were authorized to recover their full investment costs through adjustment of the electricity selling rate.

There are new nuclear projects under consideration in mature, regulated and deregulated markets that have not built a new nuclear plant for many years. In addition, emerging countries are also planning to build nuclear plants and are in the process of assessing the necessary steps to create the foundation for a nuclear energy program. In all countries the regulatory process, legislation, legal framework, and industrial and commercial infrastructure are required as the basis to go forward.

In many countries, the project models have changed from government sponsored projects, to private and in some cases public-private structures. This open market approach is driven by the reluctance of governments to incur further national debt and through promotion of deregulated energy markets. The viability of such projects hinges on the generation costs versus the acceptable market rate for selling price as well as the strength and maturity of the electricity market.

This new ownership and delivery model also has to deal with the risk associated with regulatory approval process, financing, escalation, potential cost overrun, commercial terms, national legislative support for liability, and long term power purchase agreements.

The example of these ownership and delivery structures have come forward in Finland, UK, Cernavoda 3 and 4 project in Romania and the recent request for tenders issued by the government of Turkey. These projects are focused on the involvement of private companies to finance and deliver projects that will deliver energy at market prices.

However, even in mature markets there is still expectation of government support and backstops. This has been the case in the U.S.A. with federal loan and state guarantees deemed necessary to advance the nuclear new build. In Canada Bruce Power Refurbishment project received provincial support, through guaranteed long-term power purchase agreements.

In conclusion, there is a worldwide movement shifting from 100% government or sovereign guaranteed projects towards private and public/private structures. However, these new programs are not moving forward with the speed needed to address the requirements of reducing GHGs and increased electricity demands. The high capital costs of nuclear new build requires increased equity share from the investors in particular during construction. The lenders are demanding strong power purchase guarantees and recourse to the balance sheet of the owner utilities. This coupled, with the high perception of nuclear project risks, has slowed down the new build process. At this point the role of government support in particular in the emerging markets has to be strengthened to move these programs along.

Nuclear Power for Future Electricity Generation in Ghana: Issues and Challenges

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Ghana's electricity demand has been estimated to be growing at a high rate of about 7% per annum over the last ten years. This is due to the relatively high population growth, economic aspiration of the country and the extension of electricity to rural areas.

Electricity supply, on the contrary, has been unable to meet the demand due to high dependency on rain-fed hydropower plants, which started operating in 1965 and currently account for about 68% of the total installed capacity. Within the last 28 years, climatic changes and draughts have caused the nation to experience three major power crises.

These climate changes resulted in low inflows and thus reduced power generation from hydropower systems. To complement the hydropower systems, the Government in 1997 installed thermal plants based on light Crude Oil. However, due to the high Crude Oil prices on the International Market in recent times have made the operation of these plants very expensive. Ghana's Crude Oil find can boost its energy supply when the oil exploration begins somewhere in 2010. For rural cooking, domestic biomass is employed. In addition, Ghana has no domestic coal resources. The Government of Ghana is concerned with: limited further growth potential of domestic hydro; high cost of imported oil and gas and environmental issues associated with use of imported coal.

Small Solar and wind generation exist in some sectors, but potential large-scale development is not envisioned for the near future. With the aforementioned issues in mind, the President of Ghana set up a Committee involving Stakeholder Institutions to formulate the Nuclear Power Policy and develop the basic elements of Nuclear Infrastructure and to assess the viability of introducing the nuclear power option in Ghana's energy mix. Cabinet took a decision to introduce the nuclear power option after the Committee submitted his report to the President. On 7th January 2009, there was a change of government.

There is also an IAEA TC project GHA/0/011: "Evaluating the Role of Nuclear Power in Future Options for Electricity Generation" which commenced in 2009. The question is, what are the challenges facing Ghana's Nuclear Power Programme?

Challenges and Opportunities in Launching New Nuclear Power Programs in Developing Countries

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1. Developing countries have a need to launch new nuclear power programs

As a consequence of the 1st and 2nd oil shock during 1970's, the nuclear power generation was considered the most economical energy source for more than 10 years. After that, new nuclear power programs have been showing a downward trend, due to public opinions against nuclear power programs caused by the TMI and Chernobyl Nuclear Power Plant Incidents.

However, nuclear power is coming into the spotlight these days partly because of fossil fuel shortage and the global trend of using energy resources as a diplomatic weapon. According to a recent IAEA report, 300 more nuclear power plants will be constructed in all the nations over the world by 2030. In the case of the U.S.A., the construction permits for 26 new nuclear power plants have been filed from 2007. It is considered the green light for "The Golden Years of Nuclear Energy".

Energy is one of the most important necessities to improve quality of life as well as to develop national economies. Thus, most of the developing countries put securing stable and economical energy source on their top priority to realize sustainable economic development.

I believe that launching new nuclear power programs can be a solution for securing stable and economical energy, thus I will suggest solutions to the problems of introducing new nuclear power programs in the developing countries.

2. How to solve the problems of launching new nuclear programs in developing countries?

IAEA and developed countries with nuclear energy need to help solve common challenges faced by developing countries, who intend to launch new nuclear programs.

It is required that all the international communities have to work together to conserve the environment and to control pollutions. In addition, it is a duty of IAEA and developed countries in nuclear energy to increase peaceful use of nuclear energy.

First, IAEA and developed countries in nuclear energy should provide technologies and human power training programs needed for introducing new nuclear programs in developing countries.

Though most developing countries have aspiration to get nuclear energy related technologies, it is very hard to realize their desire simply because they do not have enough resources. So, IAEA and developed countries with nuclear energy need to provide training programs to human resources; and need to give free access to a nuclear reactor for research. Also, IAEA needs to diversify attendants on its training programs; from European and American countries, to Asian countries which have strong demands for those programs. By doing so, the developing countries can reduce the time and mistakes in launching new nuclear power programs.

Second, IAEA and developed countries with nuclear energy should support financial resources to the developing countries.

The most challenging problem faced by the developing countries is raising financial resources to launch new nuclear power programs. Financial embarrassment makes the countries difficult to secure investment capital, to accumulate nuclear related technologies, and to educate human resources.

The role of IAEA supporting financial resources to the developing countries should be intensified to promote peaceful use of nuclear energy. The financial resources can be supplied by countries or companies which provide nuclear power facility to the developing countries. Additionally, IAEA needs to consider making provisions of compulsory clauses to provide financial support during the certification processes of nuclear facility design and manufacturing.

Third, IAEA should develop and deliver nuclear reactors corresponding to the needs of developing countries with adequate reactor type and capacity.

Considering the electric power grids and the capacity of each developing country, IAEA needs to develop small and medium capacity nuclear reactors. Also, IAEA should develop standardized multi-purpose (power generation, steam supply, and water conversion, etc) nuclear reactors. The SMART (System-Integrated Modular Advanced Reactor) model, a small and medium capacity nuclear reactor developed by Republic of Korea and certified by IAEA, can be a solution for the development of the multi-purpose nuclear reactor having small and medium capacity.

Finally, technological gaps among countries should be reduced in a way to motivate participation of developing countries in the international joint projects on nuclear energy. To induce developing countries to invest on the long-term research and development projects, including the 4^{th} generation nuclear reactor development program, radioactive waste and spent fuel disposal. The participation of developing countries on these projects is encouraged.

Through these measures, IAEA can achieve not only its organizational goal, the peaceful use of nuclear energy, but also establish a base for “Low Carbon Green Growth” and support sustainable development of the developing countries.

3. ROK’s exemplary model can be applied to other developing countries

For the past 50 years, Republic of Korea (ROK) has been expanding the peaceful use of nuclear energy base. Contrary to the EU and other developed countries in nuclear energy, ROK continuously accumulated skills on design, manufacturing, maintenance and operation of nuclear energy. ROK also has the know-how how to improve people’s receptiveness on nuclear energy.

ROK can be a good model for use in developing countries which intend to launch new nuclear power programs. ROK’s experience in launching and developing nuclear power programs can be effectively applied to other developing countries.

I would like to share ROK's success story and my experiences in establishing infrastructure of nuclear energy with the other member states.

Sustainability of Water Cooled Reactors – Energy Balance for Low Grade Uranium Resources

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In October 2007 the European Parliament declared, that nuclear power is indispensable for the European Union for limiting CO2 emissions. Many countries revive their nuclear power programs or start building new nuclear power plants. However, the opponents of nuclear power claim that as uranium resources get exhausted the energy needed to mine low grade uranium ore will be larger than the energy that can be obtained from fission in a nuclear power plant.

They conduct continuously their studies, publish their results in internet and present them at numerous meetings organized by antinuclear organizations. In particular they claim that the nuclear industry does not consider full energy costs of the nuclear fuel cycle, leaving aside the energy incorporated in materials and products bought from other industries and neglecting the energy needed for plant dismantling, mine area reclamation and waste management. This would result in loss of sustainability of nuclear power, with the negative energy balance expected within the next 40-60 years.

In answer to that the Institute of Atomic Energy (IAE) in Poland has performed a study of available uranium resources, energy needed for mining and milling and the CO_2 emissions in the whole uranium fuel cycle with special attention to back-end energy needs [1].

The total energy needs for uranium mining were considered, including not only electricity needed for mining and milling, for water treatment and delivery to the mine and to the neighboring settlements, but also fuel for transportation and ore crushing, explosives for rock blasting, chemicals for uranium leaching and the energy needed for mine reclamation after completed ore exploitation. In contrast to the estimates of nuclear opponents based on mining experience with rich ores mined some 30 years ago, the study of IAE has used the most up to date data, reflecting the actual state-of-art mining practices. Since the opponents state clearly that the ore containing less than 0,013% U_3O_8 cannot yield positive energy balance, the IAE study referenced three mines of decreasing ore grade: Ranger 0.234% U, Rossing 0.028% U and Trekkopje 0.00126% U, that is with ore grade below the postulated cut off value.

An exact energy balance has been made for Ranger mine, including all energy inputs, even those provided in the purchased materials. The work needed for mine reclamation has been also evaluated. It has been shown that the energy estimates of nuclear opponents are wrong for Ranger mine and go off much further for the mines with lower uranium ore grades. The study showed that the energy needed for very low grade uranium ore mining and milling increases but the overall energy balance of the nuclear fuel cycle remains strongly positive.

The reasons for erroneous reasoning of nuclear opponents have been found. Their errors arise from treating the uranium ore deposits as if their layout and properties were the same as those of uranium ore mined in the US in 70-ies. This results in an oversimplified formula, which yields large errors when the thickness of the overlayer is less than it was in the US (Fig. 1). The practice of modern mining confirms that very low grade uranium ore can be successfully mined and in fact is mined proving that the claims of nuclear opponents are in error.

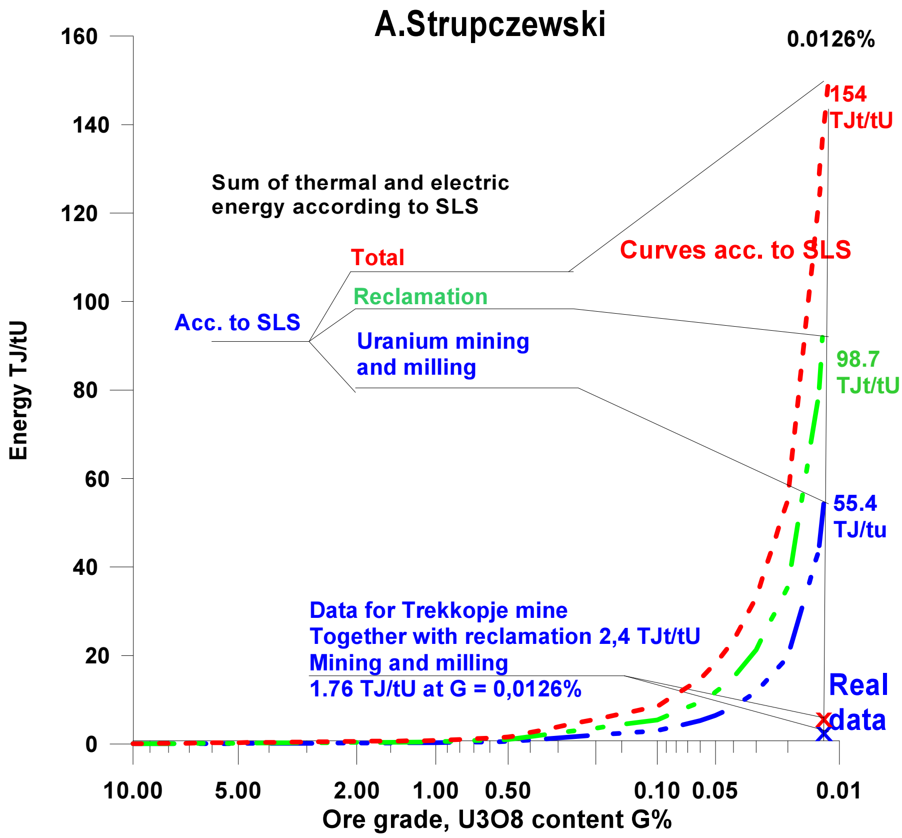

FIG. 1 Comparison of energy needs for uranium mining claimed by Storm van Leeuven and Smith with real data for low grade uranium ore mining in presently operating mines.

Further claims of high energy consumption in the nuclear fuel cycle are based on wrong estimates of the energy needed for uranium enrichment and on arbitrary assumptions concerning the energy needed for NPP dismantling and radwaste management. The paper provides the correct values based on industry experience and independent expert estimates and shows that the comparison of energy needs for various fuel cycles, with diffusion or centrifugal enrichment plants, with or without reprocessing etc.

Even for the mine using the poorest uranium ore the energy obtained at the NPP is about 70 times larger than that needed for the whole nuclear fuel cycle, including the energy needed for radioactive waste storage, the NPP construction and decommissioning to the green field status. Thus the claims of nuclear opponents are shown to be wrong [1].

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Challenges and Opportunities to Launch Nuclear Power Programme in Belarus

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Republic of Belarus, as a part of the former Soviet Union, having highly energy-intensive economy and few indigenous fuel and energy resources has been initially oriented towards nuclear power. Four nuclear power plants with total capacity of about 12 GW have been constructed near the borders of the Republic. In Belarus the constructing of nuclear CHP not far from Minsk and the planning of NPP construction in Vitebsk region have been begun. The Chernobyl NPP accident has stopped this Program.

On the other side the Republic of Belarus has been suffered from the Chernobyl accident most of all other countries including Russia and the Ukraine. About a quarter of its territory and population had turned out in the radioactively contaminated zone. The attitude of a considerable part of the Belarus population towards the nuclear energy is aggravated with the consequences of this accident.

Nevertheless recently the political decision about nuclear power development has been accepted again. It is desided to construct two units with total capacity about 2000 MW. The commissioning of the first unit is planning in 2016, the second- in 2018.

The necessity of nuclear power development has been grounded in Concept of Energy Security of the Republic of Belarus which was approved by the President of the Republic of Belarus in 2005 and in new version in 2007.

The Programme of Preparatory Works for the construction of NPP has been accepted by the Government and is under implementation.

Among another works in frameworks of the this Program it should be mentioned the following:

- site selections for nuclear facilities;
- assessment of human resource needs and availability;
- preparation for bid specification development;
- studying of nuclear technologies available and suitable for domestic application;
- developing of the Program of education and training of personal for future NPP, regulatory body and other govermental authorities, research and design institutes.

The Law "On Nuclear Energy Use" has been adopted by the Parliament of the Republic of Belarus in 2008.

The sociological monitoring of public opinion about nuclear safety of the existing NPPs and further nuclear power development was provided in Belarus in 2007-2008.

The groups of belorussian experts visited to Russia, France, China, Finland, Sweeden and Bulgaria. They studied the level of nuclear and radiation safety, national nuclear regulatory systems, technical and scientific expertise in the nuclear industry in these countries.

The Joint Institute of Nuclear and Power Research - "Sosny" was appointed as a Technical and Scientific Support Organization of Nuclear Power in Belarus.

The national regulatory body GOSATOMNADZOR has been established by decree of the President of the Republic of Belarus.

Alternatives of Financing for New Nuclear Reactors in Mexico

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Financing plays a very important role for the deployment of new nuclear reactors units in any country. Two financing alternatives can be used to support such project: the first one is that the utility provide from its own resources the capital for the investment; and the second one through international and national credits to support the nuclear project.

To be a loan candidate the viability of the nuclear project must be demonstrated, it implies among other things to have a qualified national infrastructure. Also, the utility must have an international credit record in good status by the international qualifying companies.

Both things are met by the Mexican Utility "Comision Federal de Electricidad", therefore exist the possibility to build new nuclear reactors in Mexico. Here we assess both alternatives financing and own resources projects. The conditions used in this assessment are as follows.

In the first case, there are two sources of financing, one from international credit institutions that will contribute with an 85% of the lump sum and national credit institutions that will afford the other 15%. Discount rates for the international credit according to the global market are 8%, for the national credit a 12% discount rate is used.

In this case the national scheme used is one called Financing Public Infrastructure (in Spanish Obra Pública Financiada), where the national credit institutions will support the civil works and the international credit institutions will support the Nuclear and generator islands.

Under this scheme the credit institutions or the reactor vendors through the credit support will finance the nuclear power plant construction up to be in commercial operation. In that moment the utility will start to pay the credit according to the payment schedule.

The main international credit assumptions are:

a) Payment credit period: 15 years.
b) 30 payments, each one every 6 months (does not include any payment during construction).
c) Grace period: 6 months after commercial operation.
d) Annual discount rate in dollars: 8%.

The main national credit assumptions are:

a) Payment credit period: 5 years.
e) 10 payments, each one every 6 months (does not include any payment during construction).
b) First payment: at start up of comercial operation.

c) Discount rate in dollars: 12%.

The second alternative using the utility own resources considers an opportunity resource cost with an 8% discount rate. The projects to be financed are given in Table I and Figures 1 and 2 shown the cash flow curves under each alternative.

Table I. NPP considered for financing

	ABWR	ACR	AP1000	EPR
Investment without interest (Millions of dollars)	2,031.9	1,942.6	1,411.4	2,358.7
Investment with interest (Millions of dollars)	2,399.3	2,577.2	1,790.7	2,876.6
Annual Expenditures (Millones de dólares)	183.1	191.5	152.5	239.5
Annual Income (Millions of dollars)	676.8	764.4	557.5	807.4
Generation (MWh/year)	10,359,292	11,700,745	8,533,429	12,359,169
CTNG (dollars/MWh)	35.66	33.01	34.16	37.45

Financing cash flow curve shows that the project is viable since it already considers the debt payment and in all the times for all the reactors the cash flow is positive. In the case of the use of the utility own resources it can be seen that in less than seven years the investment capital is recovered.

Both alternatives are feasible, however financing is more attractive because it allows the utility to have cash flow in any moment which does not occur when it commit its own resources.

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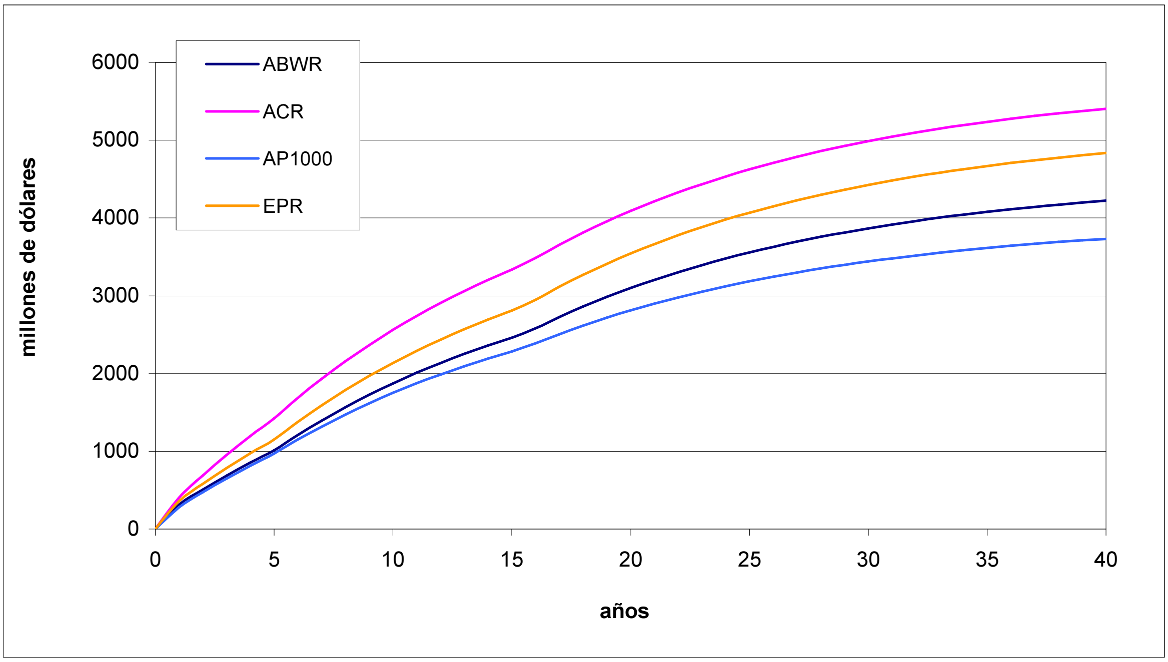

FIG. 1. Cash flow with financing.

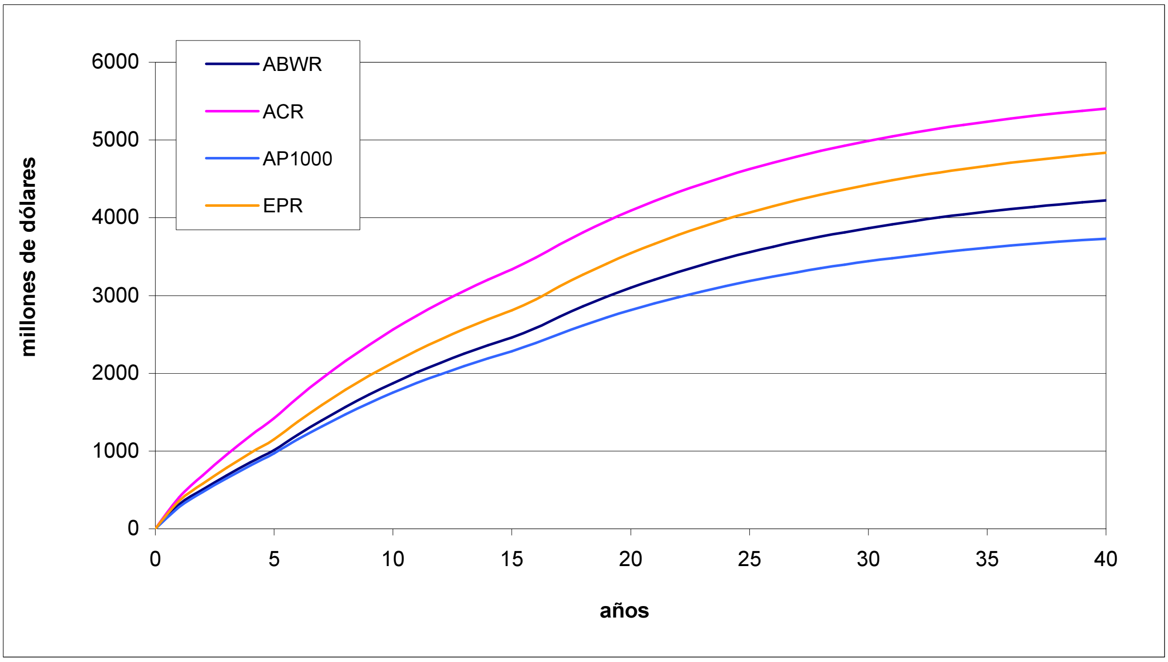

FIG. 2. Cash flow without financing.

Expert Performance Transfer: Making Knowledge Transfer Count

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INTRODUCTION

"Knowledge Transfer" is a high-priority imperative as the nuclear industry faces the combined effects of an aging workforce and economic pressures to do more with less. Knowledge Transfer is only a part of the solution to these challenges, however. The more compelling and immediate need faced by industry is ***Accomplishment Transfer***, or the transference of the *applied* knowledge necessary to assure optimal performance from experienced, high-performing staff to inexperienced staff. A great deal of industry knowledge and required performance information has been documented in the form of procedures. Often under-appreciated either as knowledge stores or as drivers of human performance, procedures, coupled with tightly-focused and effective training, are arguably the most effective influences on human and plant performance.

DESCRIPTION OF THE ACTUAL WORK

Expert Performance Transfer, or EPT, was designed to produce human performance improvement by way of procedure improvement where it counts – on-the-job. This methodology revolutionizes the way complex processes and procedures are analyzed and documented. Many complex procedures include processes that involve non-linear decision-making, an attribute of processes and procedures throughout the industry, whether in operations, maintenance, I&C, engineering, or even administration. EPT techniques are meticulously documented and are certain to improve procedure analytics; that is, the up-front analysis and decision-making that determine procedure **content**. Procedure ***content*** is a powerful driver of human and plant performance, and ultimately of efficiency and cost savings. Procedure content, presented in a coherent, logical fashion at just the right depth and breadth, can accomplish these things:

— **transfer** expert **knowledge** and technique from expert to novice performers
— **reduce** procedural **errors**, both in content and execution
— **improve** procedures **use & adherence**, since the procedure now matches the required performance

The approach to procedure content in most organizations has historically relied on assembling smart, experienced folks on a procedures team, asking them to convey their expertise in writing, and giving them a few formatting tips, along with document templates and other technology tools. Unfortunately, this approach often yields a wide variety of output; some too detailed, some not enough…with too little focus on describing the required performance that would really help the folks doing the work regardless of their level of experience. **EPT**

Methodology is revolutionary and unique in its capacity to convey expert knowledge to performers on-the-job. EPT-derived procedures contain on-the-job INFORMATION: information that is accurate, complete, and ***useful*** to workers, technicians and their managers. A procedure developed by EPT represents a superior way to achieve performance improvement because it directly affects performance WHERE it counts — on-the-job — and WHEN it counts — at the time and place performance occurs. Accordingly, guidance on the proper use of EPT has been developed using the EPT technique itself.

EPT-derived procedures are distinctive in that they:

— contain expert-level information that is accessed in REAL-TIME ON-THE-JOB
— are written at a level of detail to minimize trial and error
— reduce the amount of information recall required from memory, and
— provide directions on WHEN and HOW to perform.

EPT produces superior guidance information

The application of EPT is especially valuable when producing expert-level guidance describing howhigh complexity tasks are being performed, low frequency or unpredictable frequency tasks are performed, or during emergencies.

EPT produces procedures and guidance FORMATS based on behavior characteristics.

Fig. 1 is EPT guidance for format selection with human performance as the basis of design.

EPT Promotes Knowledge Transfer to Expert Level Considering Significant Criteria

EPT-derived procedures and guidance are a critical tool for Knowledge Transfer, enabling inexperienced workers to perform to standard, and at near-expert levels. EPT offers a documented, analytical toolkit which transcends traditional "tribal knowledge" methods of knowledge transfer, and outperforms complex but unproven knowledge management solutions based on unstructured information capture and retrieval.

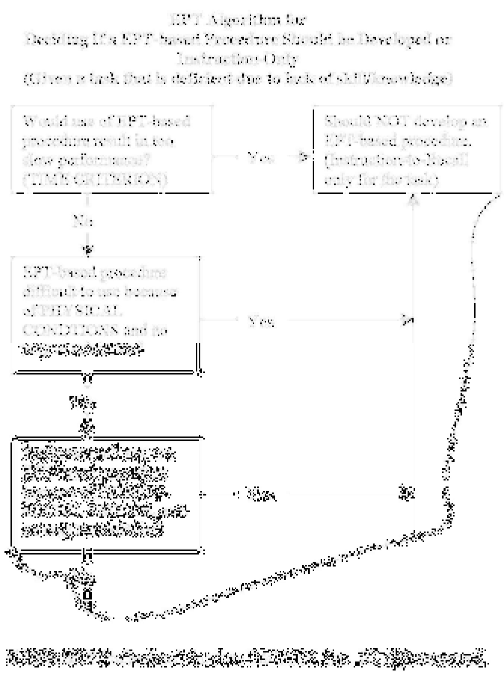

Fig. 2 is for deciding whether to develop EPT-based procedures.

EPT is a SYSTEM that itself is *SYSTEMATIC*:

- Outputs of one phase are always inputs to subsequent phases.
- Is based on a description and front-end analysis of expected on-the-job accomplishments.
- Is Derived from proven behavioral theory producing logically consistent, accountable and reproducible results.
- Decisions are rule-based with detailed algorithms and decision tables.
- Provides unambiguous guidance on amount and type of detail required to produce optimum performance.
- Designed to achieve maximum effectiveness AND efficiency.

RESULTS

EPT makes a significant difference. Contrary to a performance system relying on unaided human performance, EPT procedures do not forget information, call off sick, take vacation, or get hangovers. EPT procedures maintain proper work stimulus control during task performance, avoiding costly distractions, and they do NOT require costly retraining. Procedures are essential to effective knowledge transfer, and more importantly, ***accomplishment*** transfer. Applying EPT to develop and improve procedures results in consistently superior procedures which promote superior performance.

ACR-1000® Project – Licensing Opportunities and Challenges

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Atomic Energy of Canada Limited (AECL) has developed the Advanced CANDU Reactor®-1000 (ACR-1000®) as an evolutionary advancement of the current CANDU 6® reactor. The ACR-1000 design has evolved from AECL's in-depth knowledge of CANDU systems, components, and materials, as well as the experience and feedback received from owners and operators of CANDU plants. The ACR design retains the proven strengths and features of CANDU reactors, while incorporating innovations and state-of-the-art technology. It also features major improvements in economics, inherent safety characteristics, and performance, while retaining the proven benefits of the CANDU family of nuclear power plants.

The Canadian nuclear reactor design evolution that has reached today's stage represented by the ACR-1000 as described above, has a long history dating back to the early 1950s. In this regard, Canada is in a unique situation, shared only by a very few other countries, where original nuclear power technology has been invented and further developed.

With the Canadian nuclear technology development, in parallel, the development of Canadian regulations and licensing processes took place. This latter development was carried out by the Atomic Energy Control Board (AECB), established in 1946. The AECB mandate, originally focused on security, and later extended to include nuclear safety, was focused on regulating the nuclear sector, establishing health and safety regulations, and also played an instrumental role in forming the International Atomic Energy Agency (IAEA). The AECB, which preceded the Canadian Nuclear Safety Commission (CNSC), established in 2000, licensed all CANDU power reactors in Canada and provided assistance to the overseas regulatory authorities in licensing of CANDU reactors in Korea, Argentina, China and Romania.

The regulatory framework in Canada is currently in a period of change. The CNSC is further developing its requirements for new reactor designs in Canada whilst at the same time AECL development of the ACR design. This situation has created challenges that have been successfully overcome by both CNSC and AECL. By providing valuable feedback, AECL has worked actively in review of the existing applicable reactor regulations in Canada, and in development of new regulatory requirements and guides. In this regard, AECL provided constructive comments on several draft regulatory guides, and performed compliance self-assessment against all applicable regulations to be used with the new build in Canada including the IAEA safety requirements.

As there is no legal process in Canada for design certification of nuclear power plant (NPP), AECL initiated a pre-project regulatory review of the ACR-1000 reactor design by the CNSC to confirm compliance with regulatory requirements and also incorporate regulatory feedback in the design process to minimize project risks in obtaining future construction and operating licences for NPPs in Canada.

This pre-project review consists of two phases starting from April 1, 2008 and ending on August 30, 2009. Phase 1 ended in December 2008 and has concluded that at an overall level the ACR-1000 design intent is compliant with the CNSC regulatory requirements and meets the expectations for new nuclear power plants in Canada. This conclusion is expected to be further confirmed during the Phase 2 review that currently is ongoing. Phase 2 will go into further detail with a focus on identifying whether or not any potential fundamental barriers to licensing the design in Canada. This phase involves review of 16 topical areas:

1. Defence in Depth, Classification of Structures, Systems and Components and Regulatory Dose Limits;
2. Reactor Physics Aspects of Nuclear Design;
3. Fuel Mechanical and Thermalhydraulics Design;
4. Reactor Control System;
5. Shutdown Means;
6. Emergency and Long Term Core Cooling, Emergency Feedwater System;
7. Containment and Reactor Auxiliary Building;
8. Safety Analysis;
9. Heat Transport System Pressure Boundary;
10. Fire Protection;
11. Radiation Protection;
12. Out-of-Core Criticality;
13. Robustness, Security and Safeguards;
14. Severe Accident Prevention and Mitigation;
15. Quality Assurance in Design and Safety Analysis; and
16. Human Factors.

To ensure that the ACR design is compliant with international requirements, regulatory pre-project reviews of the ACR-700 were also conducted earlier with the CNSC and US NRC (2002-2005). The ACR-700 preceded the ACR-1000 design, and it was designed primarily for the US market. The UK regulators reviewed the ACR-1000 during the Phase 2 review conducted in the 6-month period in 2007-2008.

The regulatory feedback from these early reviews of the ACR-700 and then ACR-1000 helped AECL to better understand regulatory expectations in Canada, US and the UK to make further advancements and improvements in the ACR design to meet the Canadian and international regulatory requirements.

This paper provides an overview of the pre-project reviews by those above-mentioned regulatory bodies demonstrating opportunities and challenges in licensing process of the ACR-1000 and pointing to the importance of efficient vendor-regulator interaction.

The Role of Regulatory Authority in Licensing of the First Nuclear Power Plant

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The paper presents activity to be performed by regulatory authority in case the second approach to construction of nuclear power plant (NPP) in Poland will be started. In Poland there is some experience gathered in late 70-ties and beginning of 80-ties when the construction of the first NPP was started. Since that time the regulatory authority is active but presently limited to licensing of research reactors, national repository of radioactive waste and usage of radioactive sources in medicine and industry. Many years ago the Atomic law was established and was limited to above mentioned activities. Now the law should be appended according to requirements concerning construction of NPP. The most important field of present activity of regulatory authority should be concentrated on training of nuclear safety inspectors. It may be based on experience of still working old personnel in our office but number of inspectors is not sufficient for the new tasks. It is clear that regulatory authority should be independent of any other industry organizations and fully financed by a government. An international cooperation for regulators between countries is very important and our intentions is to establish some institutional links and follow experience from other countries.

The difference in granting licences is quite evident in many areas comparing the past time. The analysis of safety is much more sophisticated. Giving combined licence is now recommended instead of separate licenses for localization, construction, start-up and operation. The exchange of information between regulatory authorities is more general and should be intensified, especially when licensing concerns one type of water cooled reactors.

The proposed steps to be performed by Polish regulatory authority will be presented.

The Establishment of Regulation for Supporting the Development of the First Nuclear Power Plant in Indonesia

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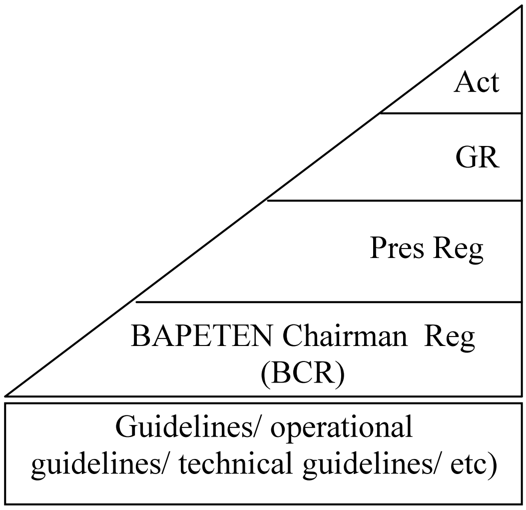
Nowadays, the government seriously establishes the management of national energy that's way on the year 2006, the government has issued the Presidential Regulation Number 5 Year 2006 on The National Energy Policy (KEN) on the period of 2025 This President Regulation stipulated that the national energy planning for nuclear energy up to the year 2025 is 2% [1] from the total national energy. To meet the need in the KEN, Nuclear Power Plant (NPP) is preparing through the nuclear energy program to be constructed four units NPP with each capacity 1000 MWe. The first NPP would hopefully be started in commercial operation by year 2017.

The national infrastructure development is needed to support the commitment of Indonesia nuclear energy program. Those includes the various aspects such as human resources development (HRD), industrial and technological aspect, management aspects, and regulatory aspects. In addition it is important in considering in the development and implementation of the nuclear program is development regulation program consistent with the International legal instruments. One of the implementation of the nuclear energy program is the establishment of independent nuclear energy regulatory authority, namely Nuclear Energy Regulatory Agency (BAPETEN). BAPETEN has function and task to control any activity using nuclear energy. There are three pillars in controlling the use of nuclear energy in Indonesia are establishing regulation, processing license and performing inspection [2]. In the establishment of the regulations of nuclear energy, BAPETEN has issued some Government Regulations and BAPETEN Chairman Regulations.

The process of the establishment of national legislation is based on the Act, namely Act Number 10 Year 2004 on the Establishment of Legislation (UU-P3). Basically, the process divided into several phases, including planning, preparation, technical preparation, formulation, discussion, approval, enactment and dissemination [3]. As the implementation of the Act No. 10 Year 2004, the Presidential Regulation No. 68 Year 2005 on the general mechanism of establishing national legal framework has been issued. This Presidential Regulation describe how mechanism of establishing and drafting of the Act. the Government Regulation as substitutes Act, Government Regulation, Presidential Regulation, and Minister/Head of Non-Department Government Agency Regulation.

The process of legal drafting to establish or to revise a regulation for controlling of the use of nuclear energy shall comply with the constitutional and institutional requirements of national political and legal system. In drafting the regulation of nuclear energy, BAPETEN is not working alone but involving other agencies or other related government agencies, and also stakeholders such as utility, academic institutions, and publics. In general, in the process of legal drafting, international publications or other country regulations can be a reference.

According to the national legislation system, BAPETEN establishes the nuclear safety regulations levels in the hierarchy of some regulations to implement the ideal nuclear program in Indonesia, namely:
1. Acts,
2. Government Regulations (GR),
3. Presidential Regulation (Pres Reg),
4. BAPETEN Chairman Regulation (BCR),
5. Guidelines / operational guidelines / technical guidelines / work instruction.
Hierarchy of nuclear safety regulations referred to above can be illustrated as following this figure:



Act is a basic legal regulation in the development and utilization of nuclear energy

GR is the implementation regulations of Act

Pres Reg is the implementation regulation of Act or GR

It is the technical regulations that implementation the Act, Government Regulation and Presidential Regulation

It is detail technical guide and non-mandatory

To provide these Act and Government Regulation operationally, BAPETEN establishes several implemented regulations in the form of BAPETEN Chairman Regulations The BAPETEN Chairman Regulation contains of more technical provisions stipulating the safety criteria on siting, design/construction, commissioning, operation and decommissioning of NPP. Some regulations are still in drafting or in planning to be formulated, At the end of the year 2010 all the regulations required for the construction & operation of the NPP will be completed issued Beside of the preparation of several NPP of regulations, BAPETEN also prepares the other regulation infrastructures, such as licensing procedures, inspection procedures, human resources development (HRD), etc. Therefore, the availability of NPP regulations and the other regulation infrastructures for NPP are expected supporting national program on construction and operation of NPP in Indonesia.

Present paper deals with the legal basis of nuclear power plant, the process of legal drafting to establish the regulations of NPP, and the current status of NPP regulations.

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The Human Resources and Training - A Mandatory Gate for Water Cooled Reactors in the 21st Century

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Despite the current situation of the global economy, the world's electricity needs will keep growing in the medium term and are about to double in 2030 from the today's figure.

As per 2007 the total worldwide nuclear electricity production was totaling more than 2700 TWHR. The renaissance of nuclear power is still ahead of us and new NPPs will from today up to 2030 amount at a total of 350 MWe installed capacity.

After the four EPR units already in construction, the projects decided or under preparation in the U.K., in the U.S. and in France for instance, confirm this steady trend.

The nuclear industry, and in particular AREVA, needs of course to adapt to this demand for new builds, and at the same time not forget to supply and serve the unabated needs of the fleets in operation.

More than 80% of our AREVA nuclear business consists in these recurring activities for the existing NPPs. Moreover our customers continue to seek for more integration of our front end, reactors and back end services. In parallel, the demand for water cooled reactors along the 21st century is foreseen to raise new challenges, industrially speaking but moreover in term of knowledge, training, and people.

In order to address this increasing demand, we are investing in all our segments of activity.

This is the case in the equipment sector where we are increasing both our primary components manufacturing capacity and our forging production capacity.

This is the case in the nuclear fuel front end with an acceleration of exploration, the opening of new mines and the construction of new conversion and enrichment factories.

This is the case in our nuclear engineering and nuclear services divisions and this is even the case in our T&D branch with continuous expansion of the industrial tool.

What does all that mean in terms of human resources? It means of course more jobs in all our business units.

From 2005 to 2008 the annual number of hiring has increased from 6000 to 12000.

Such volumes of recruitment require adapted means of integration and training of the new employees.

AREVA is managing its own specific facilities for integration and specialized training.

AREVA also contribute in Europe, the Americas and Asia, to a network of partnerships with schools and universities.

Synopses for Topic 3

DESIGN AND CONSTRUCTION OF ADVANCED WATER COOLED REACTORS

Status and Near-Term Works on the EUR Document, Possible Use by Third Parties

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For reasons (diversity of supply, evolution of the costs of alternative energies, new uses of electricity, …) that are valid in practically all the countries where they operate nuclear power plants, the major European electricity producers have expressed the will to keep the nuclear option open, that is to be able to build new Gen 3 LWR nuclear power plants when their economic interest or necessity requests it. Producing a common specification, the European Utility Requirement (EUR) document, that sets out harmonized design targets, has been one of the basic tasks towards this objective. On this base the main vendors have developed standard Gen 3 LWR designs that may be built without major design changes.

In 2007-2009, the European Utility Requirements (EUR) works have been focused on the EUR volume 3 (evaluation of the available Gen 3 designs) and volume 4 (conventional island generic requirements).

On the EUR volume 3, the assessment works on the AP1000 and AES92 projects have been concluded and the corresponding subsets of the EUR volume 3 have been published. On the EPR, since the previous assessment released in 1999 was felt a bit obsolete (both the project and the requirements had been modified since that time) the works on the EPR assessment have resumed in 2007. Representatives from ten EUR utilities and Areva have been involved in an in-depth revision of the analysis of compliance. Meetings of the specific EUR coordination group in charge of this task have been organized every 4-5 weeks throughout 2008. The revision B of the EPR subset of the EUR volume 3 has been released in July 2009. The revision C of the EUR volume 4 has been made available after a thorough review has been performed within the EUR organization to make it consistent with the revision C of the EUR volume 2 published in 2001. Meanwhile a lot of preparatory material for a possible revision D of the EUR volumes 1 and 2 has been produced since 2002. Since important contributions are not yet available the decision to integrate this revision D is still to come.

The EUR organization has kept enlarging: Energoatom from Ukraine, ENEL and Endesa have been welcomed as full members; CEZ (the Czech nuclear utility) and MVM (the operator of the Paks NPP in Hungary) are now EUR associated members. New LWR projects of potential interest for the EUR utilities are being contemplated. For instance a preliminary assessment of compliance of MHI's APWR project has been worked out in the first months of 2008.

In 2008, the EUR and ENISS organizations have decided to join their efforts in a collaboration scheme in which they coordinate their positions and actions in nuclear safety with respect to the LWR Gen 3 designs. The two organizations cooperate in their relations with the other stakeholders, in particular with the IAEA and WENRA organizations. Moreover, EUR and CORDEL (Cooperation in Reactor Design Evaluation and Licensing), which is a WNA (World Nuclear Association) working group decided also to coordinate their efforts for the industry benefit, in relation with the MDEP (Multinational Design Evaluation Program) initiative of safety nuclear regulators.

This continuous activity over more than 15 years has made the EUR organisation one of the central actors in the development of the Gen 3 LWRs in Europe and worldwide. The organisation has developed two sets of documents that are of key interest for the utilities that are considering bid processes for new NPPs. The first one is the set of generic design requirements of the volumes 1,2 and 4 that have been used several times as technical specifications in recent call for bids by utilities that did not necessarily participated to the making of the EUR document. The second set is the EUR volume 3 that gathers assessments of Gen 3 designs that have been found at an acceptable level of compliance vs. the EUR generic requirements, thus making a kind of pre-certification of a list of Gen 3 designs. Together these two sets offer an easy way to build the technical part of a call for bids: a technical specification and a short list of acceptable projects. The EUR organisation has so far given access to this documentation with limited constraints but also limited support. This policy should be continued in the coming years.

The Enhanced CANDU 6 Reactor - Generation III CANDU Medium Size Global Reactor

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The Enhanced CANDU 6^{TM} (EC6^{TM}) is a Generation III 700 class, heavy water moderated pressure tube reactor, designed to provide safe, reliable, nuclear power. The EC6^{TM} has evolved from the proven CANDU 6 plants licensed and operating in five countries (four continents) with over 150 reactor years of safe operation around the world. In recent years. this global CANDU 6 fleet, with over 92% average gross capacity factor has ranked in the world's top performing reactors. The EC6 reactor builds on this success of the CANDU 6 fleet by using the operation, experience and project feedback to upgrade the design and construction techniques. A key objective of the EC6 has been to review and incorporate design improvements in the CANDU 6 to meet current safety standards.

The key characteristics of the highly successful CANDU 6 reactor design include:

- Powered by natural Uranium;
- Ease of installation with modular, horizontal fuel channel core;
- Separate low-temperature, low-pressure moderator providing inherently passive heat sinks; Reactor vault filled with light water surrounding the core;
- Two independent safety shutdown systems;
- On-power fuelling;
- The CANDU 6 plant has a highly automated control system, with plant control computers that adjust and maintain the reactor power for plant stability (which is particularly beneficial in less developed power grids-where fluctuations occur regularly and capacities are limited).

The major improvements incorporated in the EC6 design include,

- More robust containment and increased passive features e.g., thicker walls, steel liner;
- Enhanced severe accident management with additional emergency heat removal systems;
- Improved shutdown performance for improved Large LOCA margins;
- Upgraded fire protection systems to meet current Canadian and International standards;
- Additional design features to improve environmental protection for workers and public-ALARA principle;
- Automated and unitized back-up standby power and water systems;

— Other improvements to meet higher safety goals consistent with Canadian and International standards based on PSA studies;

— Additional reactor trip coverage, based on refurbishment projects experience, to meet current Canadian Regulations.

— Both CANDU 6 and EC6 offer flexible fuel cycle options including use of slightly enriched uranium from reprocessed LWR spent fuel, high burnup MOX fuel, thorium etc in a more efficient 43 element fuel bundle carrier called CANFLEX.

— A target life up to 60 years with one mid-life refurbishment of critical equipment such as fuel channels and feeders.

— Project elements have been optimized through feedback from past construction projects. to arrive at an EC6 "in-service" schedule of 57 months from first concrete. Open-top construction method using a very-heavy-lift crane, concurrent construction, modularization and prefabrication and use of advanced computer technologies to minimize interferences are the key contributing elements for achieving this schedule.

— State-of-the-art electronic tools for engineering, safety, licensing, procurement, drahing and project management are integrated to provide complete document control during all phases of the project, including construction and commissioning. This information, in electronic format, will be turned over to the Owner for operational and configuration management needs during plant life.

— Advanced MACSTOR design for efficient dry spent fuel storage with optimized space usage.

Summary

Capitalizing on the provcn features of CANDU technology, AECL has designed the EC6 to be competitive with all forms of energy, including nuclear, while achieving high safety and performance standards.

On the Physics Design of Advanced Heavy Water Reactor (AHWR)

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The AHWR is a 920 MWth, vertical pressure tube type thorium-based reactor cooled by boiling light water and moderated by heavy water. The prime objective is to produce power utilizing thorium available abundently in India from a relatively simple system with enhanced safety level [1-4]. It is endowed with several innovative safety features such as negative coolant void reactivity, heat removal through natural circulation and passive containment cooling. The development of reactor design has drawn heavily on the experience generatred through design and operation of Pressurised Heavy Water Reactors (PHWR) and Boiling Water Reactor (BWR) in India. It was an opportunity to develop a reactor system using thorium-based fuel and gain some valuable experience. A non-proliferative thorium/U-233 based closed fuel cycle is chosen for AHWR. Plutonium discharged from PHWRs is used as the fissile seed fuel with thorium for the generation of U-233 and then as a top-up fuel in the equilibrium core along with self-sustaining U-233 in the thorium matrix. The physics design has several challenges in achieving negative void reactivity, spatial core control, on-line fuelling and minimization of inventory of plutonium fuel.

It is difficult to achieve negative coolant void coefficient in a heavy water moderated pressure tube type reactor. For this a multi-pronged approach involving pitch reduction, heterogeneous cluster design and use of mild absorbers is chosen. Plutonium bearing fuel is located separately in the outer region of the cluster with self-sustaining U-233 bearing fuel in the inner region of the cluster. A small amount of mild absorber is located in the centre of the cluster [5]. The void coefficient varies with burnup and it is a challenge to have it negative throughout the core. The state of nuclear data for the elements of interest and type of neutron spectrum in the reactor puts heavy demand on the calculation models and validation of reactivity coefficients to ensure safety [6]. A critical facility has especially been designed to carry out various lattice experiments to validate calculation models and nuclear data.

AHWR is a reactor with largely thermal spectrum and employs on-line fueling. Fuel cycle flexibility is its inherent characteristics. Equilibrium core is designed to run on self-sustaining closed fuel cycle of U-233 with plutonium discharged from PHWRs added as top-up fuel to gain in fuel burnup. Two variants of the fuel cluster with different fraction of plutonium are used to achieve self-sustaining U-233 fuel cycle. Uranium fuel does not rapidly get degraded in the closed fuel cycle in AHWR and ensuring same inventory of U-233 in the reprocessed uranium suffices to a large extent to run the next cycle. However, to arrive at equilibrium core configuration U-233 must be generated in situ. For this purpose, initial and pre-equilibrium core is loaded largely with thorium-plutonium fuel. To overcome the scarcity of fissile plutonium, uranium-plutonium fuel is also used in the initial core of AHWR.

AHWR is neutronically a large reactor in comparison to the currently operating PHWRs that makes it susceptible to xenon induced oscillations and other spatial instabilities. It is seen that the first azimuthal mode is unstable and requires spatial control and monitoring. It is difficult to provide a large number of control elements and in-core detectors at a relatively tight lattice pitch. A quadrant control scheme is chosen that matches with the thermalhydraulic design of the reactor. There are three control elements in each quarter to control the power distribution. About 150 SPNDs are installed in the core for on-line flux mapping and core monitoring. Further analysis is being carried out to ascertain the need of in-core detectors for the safety purpose.

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Advanced Construction Technologies and Further Evolution Towards New Build NPP Projects

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Hitachi, Ltd. (now Hitachi-GE Nuclear Energy, Ltd. (hereinafter, HGNE)) has constructed twenty-two nuclear power plants (NPPs) in Japan to date since 1970's, and one more ABWR plant is currently under construction. HGNE has been playing an active role in the field of nuclear power plant construction, developing and applying its advanced technologies. Included in such technologies are unique 3D-CAD-based integrated plant engineering systems and streamlined design-to-manufacturing/construction planning and management systems.

Over the last few decades, nuclear plant construction environment has dramatically changed in Japan. For instance, the number of construction workers has decreasing, while the average age of workers has become older. On the other hand, customer demands for cost reduction and for shorter construction schedule has continued to become stronger. Therefore, achieving greater rationalization in construction is one of the key issues in power plant business.

To meet these demands, HGNE has developed construction strategies based on the abundant feedbacks gained from its NPP construction experience and has made great strides in the rationalization and application of these feedbacks into its strategies. The strategies, are "1.Reduce on site work volume", "2.Leveling on-site manpower", "3.Improve on-site work efficiency", and "4.Improve on-site management work efficiency". These concpets are very simple in principle, however, their effectiveness has been absolutely proven to be huge through the successes of the past projects. In addition, HGNE believes their strategies are equally applicable to any and all power plant projects.

Utilizing all of HGNE's accumulated technologies, one of the worlds latest nuclear plants, Shika Unit 2 (Shika-2) of Hokuriku Electric Power Company (1,358MW) has been constructed "On-Budget and On-Schedule". Shika-2 was the first ABWR plant in which all the major pieces of equipment, including reactor, turbine and generator, were supplied and constructed by one prime contractor, HGNE. Also, HGNE took responsibility for the entire plant engineering from the basic design to commissioning. Shika-2 construction started with the foundation excavation of the main building in September 1999, and 58 months from rock inspection, the plant started its commercial operation in March 2006.

Now in Japan, another ABWR plant, Shimane-3, is steadily being constrcuted by HGNE in "On-Budget" and "On-schedule" manner, where HGNE is taking main role as same as in Shika-2 and more advanced design and constrcution methodolgies are being applied.

The world nuclear market are currently facing the "Nuclear Renaissance". Many NPPs are planned to be constructed in the world. We all know that the good planning and management of their "Construction" is crucial to project costs, namely, nuclear economy. HGNE believes that the application of advanced design and costruction technologies described in this paper will surely bring good results in future NPP projects all over the world.

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AP1000: The PWR Revisited

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The distinguishing features of Westinghouse's AP1000 advanced passive pressurized water reactor are highlighted. In particular, the AP1000's passive safety features are described as well as their implications for simplifying the design, construction, and operation of this design compared to currently operating plants, and significantly increasing safety margins over current plants as well. The AP1000 design specifically incorporates the knowledge acquired from the substantial accumulation of power reactor operating experience and benefits from the application of the Probabilistic Risk Assessment in the design process itself. The AP1000 design has been certified by the US Nuclear Regulatory Commission under its new rules for licensing new nuclear plants, 10 CFR Part 52, and Westinghouse has received orders for EPC in China and USA. AP 1000. The AP1000 design has also been asessed against the EUR Rev C requirements for new nuclear power plants in Europe and received certification in May 2007.

Evolution of VVER Technology towards NPP-2006 Project

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Current status of nuclear power in Russian Federation including NPP locations and reactor types is described briefly. There are 10 NPPs with 31 Units in operation, total power – 23,242 MW. In total as of 01.01.2008 the operating experience of NPPs of Russian design is equal 1030 reactor-years with the following contributions by reactor types: VVER – 350, RBMK – 490, EGP – 130, BN – 60. The chronology of commissioning the front units of each VVER-type design is given starting from the very first model as follows: 1964 – VVER-210, 1966 – VVER-70, 1969 – VVER-365, 1971 – VVER-440, 1980 – VVER-1000, 1984 – VVER-unified, 2006 – VVER-1000 modified design for China.

Every VVER reactor design version has several stages of development: 1 – technology mastering, 2 - active development of the technology, 3 – approaching the technology limits. The availability factor increasingly changes during these 3 stages, starting from about 61% to 86% and then up to 92% (the example for VVER-1000 in the period 1993 - 2008).

The technology mastering stage provides a reserve due to insufficient knowledge and imperfection . These factors are the most significant at this stage while their influence decreases further on the second and third stages.

The main requirements to the development of modern Russian nuclear energetic are as follows:

- Economic efficiency
- Guaranteed safety
- Absence of real limitations induced by the fuel base in historically significant time period
- Operating with burnt-up fuel and radioactive wastes. The fuel cycle must secure the safety of final radioactive wastes disposal
- The share of nuclear power at the country energy market shall be not less than 25 – 30%
- Structure of energy production shall provide an opportunity for nuclear energy market expansion
- The export potential: the export share in nuclear electricity generation shall be comparable to the share of its internal consumption.

The target-oriented indicators presented in the basic documents for long-term development of nuclear power industry are as follows:

№	Basic documents	Inst. Capacity, GW	Electricity generation TW-h	Load factor	NPP share
1	Strategy of nuclear power development in Russia in the first half of XXI century (NS-2000)	52	340	80%	25%
2	Energetic strategy in Russia for the period up to 2020 (ES-2003)	32/42	230/300	82%	20%
3	Federal target-oriented program “Development of nuclear power industry complex of Russia in 2007 – 2010 and outlook up to 2015 (FTP-2006)	41	300	85%	20%
4	General scheme of power plants placement up to 2020 (Genscheme -2007)	53/59	372/412	80%	23%
5	Program of Rosatom State Nuclear Energy Corporation activity for long-term period (PLA-2008)				
6	Regional federal target-oriented programs and schemes for spatial planning				

Priorities of NPPs territorial placement in Russia depend strongly on the regions involved: the European part – maximum number of sites, Siberia and Far East – isolated facilities. It should be noted that the Genscheme-2007 was adopted by Russian Government before the world economic crisis onset.

The basic project “NPP-2006” for near-term development was elaborated with the following measures taken:

– Removal of unwarranted conservatism
– Thermal scheme improvement
– Rising of steam parameters at SG outlet and reduction of pressure losses in steam lines
– Optimum use of active and passive safety systems taken from projects “NPP-91” and “NPP-92”
– Unification of major equipment
– Reduction of materials consumption

The main targets to be achieved with the new reactor designs:

– Reactor power rising
– Increasing of reactor main equipment life time
– Load factor increase
– Safety systems improvement as a means to limit occupational exposure and radioactive material release to the environment during normal operation as well as in DBA and BDBA (SA) conditions.
– Decrease of radioactive waste volume

– Exclusion the possibility of sudden large break of primary circuit by using of LBB-conception
– Maximum meeting of customer requirements

The following matters are reviewed further in the paper:

– The ways to achieve these goals.
– The main characteristics of the project “NPP-2006” in comparison with previous VVER-1000 NPP designs.
– Comparison of the modifications implemented in “Novovoronezh-2” and “Leningrad-2” NPPs projects focused on safety systems
– Characteristics of the main equipment, in particular in comparison with the previous design versions
– Status of “Novovoronezh-2” NPP construction as of August 1, 2009

Development of Next-Generation Light Water Reactor in Japan

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In Japan, the development of next-generation Light Water Reactor has been launched since April 2008. The development program will be completed in 2015. The purpose of development is to cope with the replacement for existing nuclear power plants after 2030 in Japan and the expanding demand for nuclear power in the world; "Nuclear Renaissance."

The reactor also aims to be global standard at around 2030. The requirements for global standard and domestic users have been investigated through the feasibility study of past 2 years, 2006-2007, and six innovative features or "Core-Concepts" were established as follows.

A) Reactor core system with uranium enrichment above 5% for significant decrease of spent fuel discharge and prominent higher availability

B) Long-life materials and innovative water chemistry technologies for 80 years plant lifetime and significant reduction of occupational dose

C) Seismic isolation technologies to standardize plant design independent from site conditions

D) Innovative construction techniques for significant shortening of construction period

E) The best mix of passive and active safety systems to realize economy and safety

F) Innovative digital technologies to improve availability and safety

In the first 3 years, the plant design concept and further technology development including regulatory development will be established. The applicability of the technologies to the nextgeneration LWR will be also investigated. Screening tests for innovative materials for fuel cladding, reactor core internals and steam generator tube have been conducted. The plant performance, such as safety, reliability, capacity factor, will be evaluated quantitatively.

In the first quarter of 2010, the whole development program will be revalued based on the above tests and engineering activities and then the developments with large-scale and long-term tests will be started.

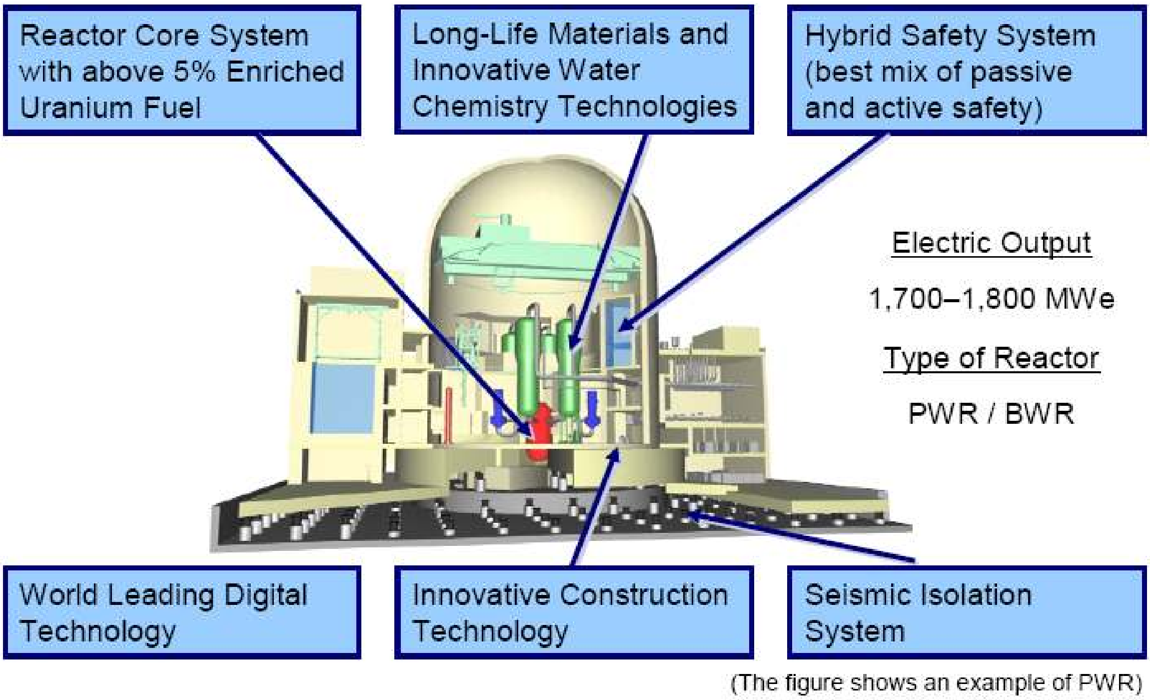

FIG. 1. Six-concepts of the next-generation LWR

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Open Issues Associated with Passive Safety Systems Reliability Assessment

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The efforts conducted so far to deal with and evaluate the passive safety systems reliability to be implemented in advanced water cooled reactor designs [1], has aroused an amount of open issues to be addressed in a consistent way, in order to endorse the proposed approaches and to add credit to the underlying models and the eventual reliability figures, resulting from their application.

In particular, with reference to the so-called t-h (thermal-hydraulic, i.e. resting on natural circulation) passive systems, this concerns, for instance, the choice of the distributions to be assigned to the relevant parameters (i.e. the parameter probabilistic characterization) [2], as input to the related analysis.

From the analysis of the various methodologies, which have been developed over these most recent years within the community of the safety research, and are currently available in the open literature, the following open questions are highlighted and consequently needs for research in all related areas are pointed out:

— The aspects relative to the assessment of the uncertainties related to passive system performance: they regard both the best estimate t-h codes used for their evaluation and system reliability assessment itself;

— The dependencies among the parameters, mostly t-h parameters, playing a key role in the whole process assessment;

— The integration of the passive systems within an accident sequence in combination with active systems and human actions;

— The consideration for the physical process and involved physical quantities dependence upon time, implying, for instance, the development of dynamic event tree to incorporate the interactions between the physical parameter evolution and the state of the system and/or the transition of the system from one state to another

— The comparison between active and passive systems, mainly on a functional viewpoint.

Focus on these issues is very important since it is the major goal of the international research activities (e.g. IAEA) to strive to reach a common consensus about the different approaches.

All these points are presented and discussed and a viable path towards the implementation of the research efforts is delineated as well.

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Design Characteristics of the Advanced Power Reactor 1400

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I. INTRODUCTION

The APR1400 is an evolutionary ALWR for which the design is based on the current KSNP design with capacity evolution. It also incorporates a number of design modifications and improvements to meet the utility's needs for enhanced safety and economic goals and to address the new licensing issues such as mitigation of severe accidents. To establish the safety and economic goals for the APR1400 the requirement for ALWRs was compared thoroughly through the safety and economic evaluation. The design requirements have been established based on this comparative study, and the major requirements for the APR1400 design are:

- General Requirements
  - Capacity: 4000 MWt (rated thermal power);
  - Plant lifetime: 60 years;
  - Seismic design: SSE 0.3g;
  - Safety goals: core damage frequency lower than 10E-5/RY; and frequency of large radiation release due to containment failure lower than 10E-6/RY.
- Performance Requirements and Economic Goals
  - Plant availability: 90%;
  - Occupational radiation exposure: less than 1 manSv per reactor-year;
  - Construction period: 48 months for Nth Plants; and
  - Economic goal: 20% cost advantages over competitive energy sources.

As noticed above, the APR1400 aims at both enhanced safety and economic competitiveness. The economic goal of APR1400 is considered achievable by high performance in operation and cost savings in construction.

II. DESCRIPTION OF THE NUCLEAR SYSTEMS

The nuclear steam supply system is designed to operate at rated thermal output of 4000 MW to produce an electric power output of around 1450 MWe. The major components of the primary circuit are a reactor vessel, two coolant loops, each containing one hot leg, two cold legs, one steam generator (SG), and two reactor coolant pumps (RCPs), and one pressurizer (PZR) connected to one of the hot legs. Two SGs and four RCPs are arranged symmetrically. The design temperature in the hot leg is reduced from 621˚F of OPR1000 to 615˚F in order to increase the operating margin and to reduce the SG tube corrosion problem. The capacities of the PZR and the SGs (especially secondary side) are increased from that of current designs. The increased capacity of the pressurizer accomodates the plant transients without reactor trip up to Condition III transients.

Conventional spring loaded safety valves mounted to the top of the PZR are replaced by the pilot operated safety relief valves (POSRVs), and functions of the RCS overpressure protection and safety depressurization could be performed by the POSRVs.

The power control system is capable of daily load follow operation at a typical load variation profile in Korea; 16 hours at 100 % and 4 hours at 50% with 2 hours ramps for power decreases and increases. The load rejection capability at the rated power should also be incorporated. This capability can reduce the outage time caused by the secondary system troubles since the reactor power can be brought up to 100% as soon as the troubles have been fixed.

III. SAFETY CONCEPT

One of the APR1400 development policies is to increase the level of safety dramatically. To implement this policy, the plant has been designed in accordance with the established licensing design basis to meet the licensing criteria and also be designed with an additional safety margin in order to improve the protection of the investment, as well as the protection of the public health.

The safety goals of the APR1400 can be summarized as follows;

— The total core damage frequency should not exceed 10E-5 per year, considering both internal and external initiating events.

— The whole body dose for a person at the site boundary should not exceed 0.01 Sv (1 rem) during 24 hours after initiation of core damage with containment failure. The probability exceeding such a limit should be less than 10E-6 per year.

— The frequency of an accident in which the release of long-lived radioisotopes such as Cs-137 would exceed the amount to limit the land use shall be less than 10E-6 per year.

In addition to the public safety, a concept of investment protection has been implemented. In APR1400, there are many investment protection goals such as loss-of-coolant-accident (LOCA) protection, steam generator inventory, and so on. For example, the reactor with its fuel should be used continuously following the event of small break LOCA up to 15 cm pipe break.

Another important design philosophy for safety is the increased design margins. A few examples of the design requirements following this philosophy are the requested core thermal margin of 10~15%, sufficient system capacity for operator recovery action time of more than 30 minutes, and station blackout coping time of 8 hours.

IV. CONCLUSIONS

The APR1400 design development started in 1992. The basic design was completed in 1999. Since then, we performed design optimization to enhance economics. Through design optimization process, current APR1400 design has been finalized.

From 2000, Korean regulation authority conducted the safety review for design certification and awarded design certification at May, 2002. Currently, the project for constructing first commercial APR1400 nuclear power plants, namely Shin-Kori 3&4 is in progress. Currently, the commercial operation of the first APR1400 is planned in 2013.

Synopses for Topic 4

SAFETY AND PERFORMANCE ACHIEVEMENT IN CURRENT NPPS

Plant Life Management Activities for Long Term Operation of the Argentinean Water Cooled Reactors

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The Comisión Nacional de Energía Atómica (CNEA) –National Atomic Energy Agency of Argentina– is a Sate-owned Research and Development (R&D) institution that has among its functions the responsibility of keeping up to date, and available to the Utilities, all nuclear related technologies in order to ensure the highest performance of the plants in terms of safety and production. In 2005 CNEA and Nucleoeléctrica Argentina Sociedad Anónima (Na-Sa) –Argentinean National Utility– have formed a joint working group to develop Plant Life Extension (PLEX) methodologies to be applied in the Argentinean CANDU-6 plant "Central Nuclear Embalse" Refurbishment Project. Training and supervision have been provided by Atomic Energy of Canada Ltd. (AECL). As a result, a solid group of engineers is currently working in the PLEX division of the plant, and has finished the first stage of the project that consists in evaluating the current condition of the major equipment of the plant, in order to make the business case for the refurbishment. In the other hand, CNEA has continued to develop its own group to cope with all Long Term Operation (LTO) management issues. Several experimental activities are being carried out in the corrosion and cable degradation areas; the emphasis is put on the prediction of the future behaviour of the materials, based on their current condition and on accelerated ageing tests. It is planned for the near future to set up a Loss of Coolant Accident (LOCA) test facility to study the behaviour of the materials in LOCA and post-LOCA conditions. It is worth mentioning that one of the Team objectives is being achieved throughout these activities; where the experimental facilities and expertise of the Research & Development Institution are combined with the operational and In-service experience from the plant personnel. From the methodological point of view, the PLIM-PLEX division of the Comisión Nacional de Energía Atómica is currently developing an unified PLIM program for the Long Term Operation of all current and future Argentinean reactors. The experience gained during the Embalse Refurbishment Project is to be applied and expanded to design a single and comprehensive program. Proper procedures are being developed for each part of the unified program that can be divided into the following stages:

Design Review and Screening of major System, Structures and Components (SSCs): In the first part of the program; it is necessary to fully understand the design bases and operational conditions of the plant, in order to determine the major components/group of components that are going to be covered by the LTO program. Once the scope of the program is established, selected SCCs should be grouped to set up specific Ageing Management Programs (AMPs) for each SSC.

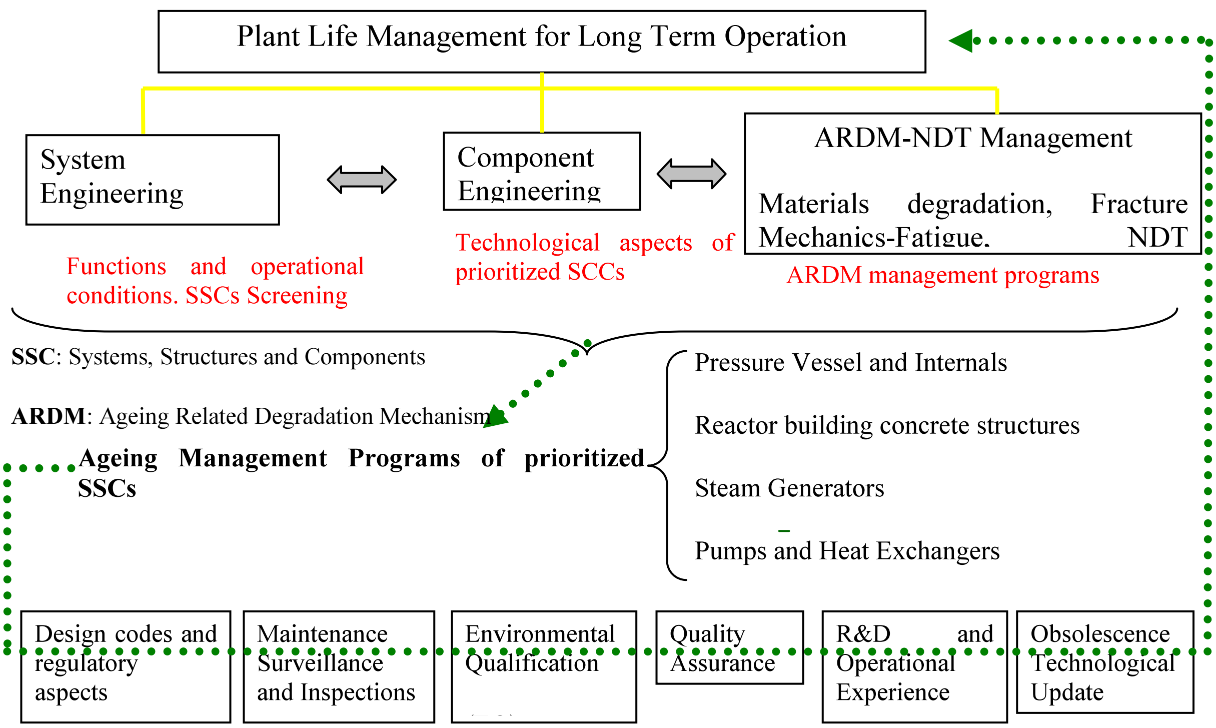
Development of a Generic Ageing Database: The objective of this stage is to have a single and periodically updated database of the recommended Maintenance, Surveillance and Inspections (MSI) practices within the different AMPs. This will allow for a continuous revalidation of the Plant Programs. As this regard, CNEA is participating in the IAEA activities related to the International Generic Ageing Lessons Learned Database.

Analysis and /or design of specific AMPs: In this part of the program, detailed procedures for the MSI activities included in each AMP have to be established. It is important to point out the different strategies to be followed when AMPs for a new reactor are designed or existing plant programs for a working reactor are re-assessed. However, in any case, the adequacy and fulfillment of the programs has to be compared against international standards and good practices (summarized in the previous stage - Generic Ageing Database)

Determination of measurable Program Performance Indicators: For each AMP it is essential to establish proper indicators allowing for a periodic evaluation of the effectiveness of the program. Temperature values, pressure drops, bearing replacement rates, heat exchangers tube plugging rates and corrosion rates, among others, are parameters that could be followed to evaluate the AMP effectiveness. The selection of proper indicators requires a case by case analysis and strongly depends on the particularities of each AMP

AMP evaluation and feedback: Not only the selected indicators are to be evaluated, but also field experience and any other relevant data have to be collected and analyzed in order to improve the effectiveness of the AMPs. The permanent re-evaluation of the program allows for an optimization of the resources assigned to the MSI activities of each AMP

The following chart summarizes the interactions between the different areas of the plant, needed for the proper establishment of a Plant Life Management for the Long Term Operation of Argentinean water cooled reactors.



Powering the Future - Life Extension of the Point Lepreau Generating Station

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The Point Lepreau Generation Station is a CANDU 6 PHWR design that went into service in 1982 in the province of New Brunswick, Canada. It operated with a lifetime capacity factor of 82% until it was shut down in March 2008 to commence a Refurbishment Outage to extend the life of the station another 25 – 30 years. This paper describes the challenges facing the station late in its life, the assessment of feasibility for life extension, and execution of the Refurbishment Outage.

Early its life, the station routinely achieved > 90% annual capacity factor. Operational issues with fuel channels and feeders beginning in the mid 1990s were the prime reasons for the reduction in capacity factor.

> To mitigate the impact of Pressure Tube diametral creep on reduced safety margins to fuel dryout, reactor power was reduced over time. By 2008, reactor power had been reduced by 12%.
>
> Two long outages (3 to 6 months) were required to relocate the fuel channel spacers to their design locations to prevent contact between the pressure tubes and the calandria tubes. Contact is undesirable because it promotes the concentration of zirconium deuteride in the contact area making the tubes susceptible to brittle failure.
>
> Feeders experienced FAC thinning and SCC cracking at the tight radius bends. This resulted in two forced outages because of through wall cracks and longer planned outages to inspect the “at risk” feeder bends and to replace those bends where cracks were detected or those that had insufficient wall thickness margin for the next operating cycle.

Replacement of the fuel channels and feeders would resolve the above issues. A comprehensive Condition Assessment process of the stations structures, systems and components was conducted to determine the other issues that would have to be addressed to life extend the station. An Integrated Safety Review was done based on IAEA NS-R-1, IAEA NS-G-2.10 and CNSC RD-360 (draft) to determine gaps with international Safety Goals, modern codes and standards and regulatory requirements. The outputs from these analyses determined the scope of a Refurbishment Outage. In July 2005, approval was given to Refurbish the station commencing in 2008. Expected capacity factor during the extended life is 89%.

The Refurbishment Outage has three phases:

— Station shutdown, defuelling and dewatering

— Execute the modifications, replacements and repairs

— Commission and return to service

The first phase was completed on schedule in May 2008. The reactor core was defuelled using the Fuelling Machines (12 bundles in each of 380 channels). Following defuelling, heavy water was drained from the heat transport and moderator systems and a vacuum drying system installed to reduce the tritium hazard. In addition, the secondary side of the steam generators were cleaned and other water systems were laid-up. The local air coolers in the vicinity of the reactor faces were removed to provide space for the reactor fuel channel replacements.

The second phase is well underway. The critical path is through the replacement of the fuel channels and the feeders. Other important work includes:

— Replacement of shut down system computers

— Improvemets in shutdown trip coverage

— Improvements in Severe Core Damage and Large Release Frequencies

— Address regulatory issues

— Primary side cleaning of the steam generators

— Generator overhaul and turbine uprate

Replacing the fuel channels and feeders consists of the following steps: remove feeders, remove end fittings, remove pressure tubes, remove calandria tubes, inspect the internals of the reactor, install calandria tubes, install pressure tubes, install end fittings and install feeders.

Tooling designed to shear the feeders did not perform as expected; consequently, the feeders were removed manually. This presented contamination problems that were difficult to resolve.

Custom-designed, automated electro-mechanical tooling was required for the removal of the fuel channel components. Local trades were trained to operate the tooling. At the time of the writing of this synopsis, pressure tube removal is 90% complete. The processes to remove fuel channel assemblies are complex to limit manual activities to keep the radiation exposure low. The production rates of the tools have been less than predicted largely because of the precision required to engage the components and the impact on the mechanisms from dust and debris from the cutting and shearing operations. More detail on the fuel channel replacement program will be presented in the final paper.

Despite the equipment problems, actual radiation exposure is tracking below predicted.

R&D Activities for NPP Life Management in Korea

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Long term operation (LTO) of a nuclear power plant (NPP) becomes the recent worldwide trend because of unstable oil price and the green house effect. To make LTO possible, first of all the plant safety should be maintained in its extended operation period and then the economical benefits should be also expected. For the plant safety, it can be confirmed through periodic safety review (PSR) system recommended by International Atomic Energy Agency (IAEA). The economical benefits may be achieved by the plant life management (PLiM) as well as structural integrity of the critical components of NPP.

Systems, structures and components (SSC) of NPP are designed to have safety margins in design stage, and being operated with operational margins. As the plant gets older and older, however, aging of SSC occurs and some of SSC may be sometimes failed due to the unexpected aging mechanisms in design stage. Most countries which have operated NPP have their own R&D programs to establish proper countermeasures against the aging and degradation of SSC. The well known programs are materials reliability program (MRP) and steam generator management program (SGMP) of EPRI.

Many R&D activities related to PLiM have been carried out and implemented to Korean NPP for the long term operation beyond their original design lives. Those activities include PLiM study, PSR, SGMP, MRP, thinned pipe management program, study on dissimilar metal welds, aging monitor etc.

PLiM studies for PWR and PHWR have been completed. Detailed lives for critical NPP components were evaluated and aging management programs (AMP) were established for LTO of Kori Unit 1 and Wolsong Unit 1 in particular.

Korean SGMP has been also developed for operation and maintenance of steam generator based on its performance criteria and implemented to all PWR in Korea. In the work, particularly, the retired steam generator from Kori Unit 1 in 1998 was very usefully utilized to develop robot for withdrawal of tubes, to improve the detecting technology by using its natural defects. A program for integrity assessment of steam generator tube, PIAT®, was developed by KEPRI, which is a comprehensive wear assessment program including thermal-hydaulic data base, mode analysis, flow-induced vibration and wear assessment.

MRP has been developing at present. In the project, AMP for nickel-alloy nozzles and penetrations was prepared to ensure their integrity against PWSCC in PWR. The AMP includes the survey of all Alloy 600 components in Korean NPPs and the assessment of integrity of components highly susceptible to PWSCC. Pressure tubes and feeder pipes are the most critical components in CANDU reactor. AMP for those critical components will be established.

Thinned pipe management program has been also developed and implemented to all NPP in Korea. An integrity assessment criterion and a computer program (PiTEP®, Pipe Thinning Evaluation Program) for wall thinned piping items were developed by KEPRI for the first time in the world, which is directly applicable to the secondary piping system of nuclear power plant.

Study on dissimilar metal weld has been also developing at present. Research to develop the technologies for analysis and measurement of the residual stresses induced by butt welding is on going. A guideline for residual stresses analysis in dissimilar metal welds was developed by using finite element methods. To obtain the reliable analysis result for actual dissimilar metal weld of pressurizer nozzle, round robin analyses with 5 participants were performed. R&D project on comprehensive assessment and maintenance technologies for the dissimilar metal welds will be started sooner or later.

A government-supported project, “Development of Aging Monitor for Operating NPP Components(2007.4~2010.2)”, has started to develop aging monitor which can display on-line aging mechanisms and states of various NPP components. At present, aging monitors were developed for OPR-1000, CANDU type reactor and Westinghouse 3-loop type reactor. Aging monitors for Framatome type and Westinghouse 2-loop type reactor will be developed.

With the results of R&D activities, continued operation of Kori Unit 1 was successfully started from January 17, 2008 for next 10 years beyond its design life. It must be a landmark of 30 years history of nuclear power generation in Korea. Subsequently Wolsong Unit 1 is also expected to start its continued operation when the replacement of pressure tubes and feeders are completed in 2009.

In this paper, all the activities and their results of the R&D programs will be introduced.

Nuclear Power Plant Life Management - Challenges and Proposals for A Unified Model Integrating Safety and Economics

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The current social and economic framework for the energy production is characterised by the following trends:

the open electricity market, which demands for significant reduction of the generation costs, very strict investment planning, outsourcing, controlled reliability of the equipment and components (incl. obsolescensce) and therefore for reliable indicators of the effectiveness of the maintenance programmes

The generic trend towards the extension of the operating life of the existing plants, which requires a detailed review of the original design assumptions, also reflected into current maintenance practice, and the continuous monitoring of the component reliability (performance goals) in order to support a suitable trend of the safety evaluation beyond the design life.

As a consequence, in last years many electric utilities and nuclear power plants adopted policies for improved coordination of both safety and non-safety programs, called plant life management (PLIM).

In Countries with some experience, the PLIM program proved very convenient, especially when coupled with Maintenance, Surveillance an Inspection (MS&I) optimization: average savings are reported in the range of 20-30% of total (maintenance) costs.

Moreover, in terms of safety, the control of equipment reliability, significantly improved with PLIM models for example through Ageing Management Program (AMP) and Reliability Centred Maintenance (RCM), made a long term asset management of the overall plant possible and the overall safety indicators significantly improved in many cases.

This is why R&D tasks are needed in this phase, not only in the long term extrapolation of the component integrity and behaviour, but a also in new management strategies at the plant (PLIM), able to address organisational issues, spare part management, staff ageing, component obsolescence, etc, which are typical components of the PLIM.

The Framework Programme 7 of the EU in the area dedicated to the reactor systems calls for a research effort “to underpin the continued safe operation of all relevant types of existing reactor systems (including fuel cycle facilities), taking into account new challenges such as life-time extension and development of new advanced safety assessment methodologies (both the technical and human element)”.

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The need for harmonised PLIM models at least at the European level (where the energy market is rather unified and regulated by general rules) has been recently raised, supported by two main reasons:

A reliable PLIM model needs to be supported by a consistent analysis of feedback from plant operation, which could be available only at over-national scale (only exceptions are the large nuclear energy suppliers);

Effective PLIM models need sharing of resources, suppliers, spare parts, O&M techniques among different plants, which therefore must have the same or very similar characteristics to foster such exchanges.

The program structure described in the paper is shown in Figure 1, where the integration of the existing programs at the plant is highlighted.

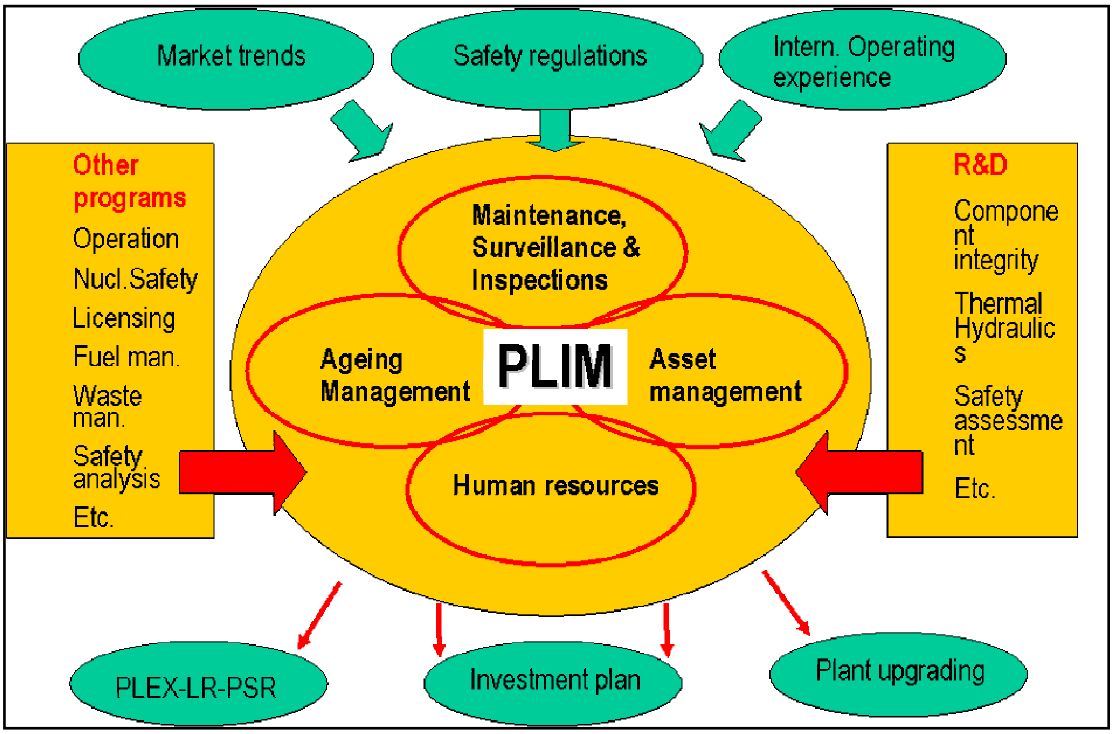

Figure 1 – Proposed structure for a unified PLIM program

This paper provides a short summary of selected results obtained at the JRC/IE in the development of a UE model for PLIM, explicitly addressing program management issues (organisation, contractor management, and program indicators), maintenance optimisation issues, ageing management of selected components and structures, and human reliability issues, as suggested by the review of some examples of PLIM programs in the EU.

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Long-Term Aging Management Strategies for Nuclear Power Plants

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All nuclear power plants have implemented some form of long-term aging management at their facility. However, methodologies have been developed recently that can provide enhancements to the existing aging management plans. This paper presents these comprehensive aging management strategies.

Comprehensive long-term aging management strategies consider the role of every system, structure, and component (SSC) in the nuclear power plant and assesses how aging can prevent those SSCs from performing those roles. Nuclear power plants that have not implemented comprehensive long-term aging management strategies/programs utilize existing maintenance and inspection programs to manage plant aging. While maintenance and inspection have not been traditionally called aging management programs, they perform activities that are credited with managing aging. These activities have been performed for many years, and many of these programs exist as a result of commitments to specific regulations (e.g. In-Service Inspection). Other programs exist because they have been identified by standards or industry best practices. Operating experience provides feedback when failures occur so that enhancements to programs can be developed through the corrective actions process.

While the above process has served the industry well and will continue, many gaps have been identified in the processes of plants that have implemented long-term aging management strategies. These gaps include, for example, missing programs, key components that are experiencing aging but aren't covered by any program, deficient or out-dated techniques being used in existing programs, and enhancements in the acceptance criteria for the program.

Several methodologies have been developed to identify gaps in aging management and/or to confirm that the existing aging management is sufficient. These methodologies combine to provide a comprehensive long-term aging management strategy. While the ultimate goal of these methodologies is the same (that is, managing the aging of plant SSCs to assure safety and cost efficiency) their focus and stakeholders are different, and therefore, there are differences in the specific methodologies. These categories of methodologies can be defined as regulatory-based, economic-based, and experience-based methodologies. This paper will provide an overview of the three different categories of aging management methodologies.

Regulatory-Based Aging Management

While all aging management is based on experience, there have been many regulations specifically developed to manage component aging. Aging management that is based on regulatory requirements is generally focused on ensuring the public health and safety, and therefore, is focused on safety-related and important to safety system, structure and component functions. Examples are requirements related to ensuring electrical components are qualified to operate in the post-accident environment (Environmental Qualification) and rules which ensure that the reasctor vessel will not experience unacceptable loads due to thermal shock events (Pressurized Thermal Shock). These regulatory requirements implemented procedural requirements into plant programs for performing specific actions related to plant surveillance, operation, or maintenance.

In the more recent past, additional regulatory requirements have been established in some countries that broadly cover aging management. These requirements are for 'oversight' programs which implement criteria for identifying the scope of aging management programs, while providing specific evaluation methodologies and/or criteria for assessment of plant SSCs. In the United States, these regulations include the Maintenance Rule and the License Renewal Rule. While the short-term goal of the License Renewal Rule is for plants in the US to obtain a renewed operating license, the purpose of the methodologies supporting the regulatory requirements is to implement long-term aging management programs.

With respect to aging management, the International Atomic Energy Agency (IAEA) Periodic Safety Review (PSR) focus is the same as the aging management focus for US plants that have received renewed operating licenses. The PSR rationale and objective is that the member utilities evaluate the cumulative effects of aging on plant SSCs, and that each SSC should be assessed against its design basis to confirm that aging has not significantly undermined the design basis assumptions.

Economic-Based Aging Management

While regulatory requirements focus on the aging management that is necessary to ensure public health and safety, the focus of economic-based aging management methodologies is the efficient operation of the power plant. While aging assessments for SSCs in this category of the plant are similar or the same as for the first category above, the consequences of the SSC aging may not be the same. For example, while proactive aging management of turbine components may not have any consequence to the health and safety of the public because their failure cannot impact safety or important to safety functions, failure of these components may have significant impact on plant availability and economic viability. Therefore, the economic-based aging management methodologies evaluate SSC aging against their importance to power production or other related economic criteria.

The primary difference between the methodologies for economic-based aging management and regulatory-based aging management is consideration of the economic impact of the consequences of the aging management, or lack of aging management being performed. For major components that impact power production (turbine, major heat exchangers, etc.) evaluations are performed using operating experience to predict component performance. These evaluations are then assessed and integrated with overall utility strategic investment objectives to optimize long-term planning.

Experience-Based Aging Management

The final methodology used for managing aging is the experience-based methodology. This method is currently being used throughout the industry and has been employed since the birth of the nuclear industry. This method relies on operating experience, owners groups, industry message boards, and similar methods to disseminate information related to experience on issues related to component aging. While this category of aging management is not a single specific aging management methodology, the various methods of communication throughout the industry have been very effective at communicating and managing issues over the years.

The methods employed in experience-based aging management ranges from informal information sharing to formal industry groups. Examples of these methods range from the informal sharing of information between system engineers that have the same system in different plants to the formal organization of the Boiling Water Reactor Vessel Inspection Program (VIP) or Pressurized Water Reactor Materials Reliability Program (MRP).

This paper will present a general overview of each of the categories, including examples of results of implementing and the consequences of not implementing the categories of methodologies.

Innovative Maintenance Technology for RPV and RIN of Operating Nuclear Plant

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Considering present nuclear operating power plants, the number of long term operated plants that are being operated more than 30 years has increased in recent years. There are many maintenance technologies developped and applied for the purpose of mitigating, improving or repairing these operated plants.

In this paper, at first, ordinary maintenance sequences being applied to reactor pressure vessel and reactor internals of Japanese operating nuclear plants will be introduced with related maintenance technologies classification. And out of many maintenance technologies, some characteristic ones will be explained.

Maintenance technologies using laser power have been developped to mitigate or repair SCC of reactor pressure vessels and reactor internals.

Laser peening is the technology to improve surface residual stresses to mitigate SCC. This paper will show recent development of this technology, and in addition to this, some consideration or information related to this technology will be explained. That is to say, considering certain external load, knowledge of relaxation of improved stress formed by laser peening will be explained. And also, consideration to be studied before laser peening application on the weld line that already includes some existing SCC area will be recommended.

Underwater laser beam welding technology is another laser application technology, that has been developped for the purpose of mitigation and repair SCC. The recent development of this technology will be shown in this paper.

And one more technogy, integrated reactor internal component replacement work for BWR, will be presented in this paper. In Japan, it has already been performed for several times. Through this replacement work, very large area of welded type reactor internals, such as core shroud or jet pump can be replaced at one time to prevent or repair SCC.

Detail information will be mentioned in the paper.

IAEA Surveillance Data Administration within Mat-DB

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JRC-IE has developed a web-enabled materials database (Mat-DB) for storing materials test data resulting from international research projects together with other documentation in a related document management database (DoMa) providing fast access to confidential and public data sets. The databases are implemented in the secure ODIN portal: **https://odin.jrc.ec.europa.eu** of JRC-IE.

Mat-DB covers thermo-mechanical and thermo-physical properties data of engineering alloys at low, elevated and high temperatures for base materials and joints, including irradiated materials for nu-clear fission and fusion applications, thermal barrier coated materials for gas turbines and properties of corroded materials. The corrosion part refers to weight gain/loss data of high temperature exposed engineering alloys and ceramic materials. For each test type the database structure reflects international test standards and recommendations.

Mat-DB features an extensive library of evaluation programs for web-enabled assessment of uni-axial creep, fatigue, crack growth and high temperature corrosion properties. Evaluations can be performed after data retrieval or independently of Mat-DB by transferring other materials data in a given format to the programs. The fast evaluation processes help the user to get a detailed data analysis or data extrapolation for component design and life-time prediction.

Within an bi-lateral agreement Mat-DB hosts also thousands IAEA surveillance data of reactor pressure vessel materials from different nuclear power plants of their member states. These tensile and impact materials data tested before and after irradiation are uploaded in a confidential area of Mat-DB only accessible for the entitled IAEA administrators. The paper provides an overview over:

- reflection on the complexity of IAEA data transfer into Mat-DB;
- implementation of standardized XML technology to ease complex data exchange in the future,
- security, access rights and data confidentiality,
- advantage of on-line data access by nuclear power plant members world-wide,
- data evaluation,

— maintenance and upgrades of Mat-DB,

— IAEA surveillance data content within Mat-DB,

— continuous uploading of new IAEA surveillance data, advantages in using the web-enabled Mat-DB for IAEA members.

The Best-Estimate Plus Uncertainty (BEPU) Challenge in the Licensing of Current Generation of Reactors

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Among the general attributes of a methodology to perform accident analysis of a nuclear power plant for licensing purposes, the very first one should be the compliance with the established regulatory requirements.

A second attribute deals with the adequacy and the completeness of the selected spectrum of events which should consider the combined contributions of deterministic and probabilistic methods.

The third attribute is connected with the availability of qualified tools and analytical procedures suitable for the analysis of accident conditions envisaged in the concerned Nuclear Power Plant (NPP). Thus, a modern and technically consistent approach has been built upon best estimate methods including an evaluation of the uncertainty in the calculated results (Best Estimate Plus Uncertainties or BEPU approach).

The complexity of a NPP and of the accident scenarios may put a challenge for a conservative analysis and may justify the choice for a BEPU approach in the licensing process. This implies two main needs: the need to adopt and to prove (to the regulatory authority) an adequate quality for the computational tools and the need for the uncertainty.

The purpose of the present paper is to outline key aspects of the BEPU process aimed at the licensing of the Atucha II NPP in Argentina. The Atucha II is a heavy-water cooled heavy-water moderated, vessel type, pressurized reactor. The moderator fluid has the same pressure as the coolant fluid, but temperature is lower. Fuel channels, which do not withstand pressure difference during nominal operation, separate the coolant from the moderator. The thermal power produced in the moderator is used to pre-heat the feed-water.

A direct link with the bases of nuclear reactor safety shall be ensured by the 'BEPU-description document'. In the present case this is formed by the following main elements or steps:

1) Evaluation of the possibility to use a BE estimate within the context of the current national (i.e. of the Country where the NPP is installed) Regulatory Authority (RA) requirements. A pre-application document was submitted to the national RA. This included the consideration of past interactions between the RA and the applicant as

well as the analysis of the licensing practice in the Country where the NPP was designed.

2) Outline of international practices relevant for the proposed approach. The experiences acquired in the use of Best Estimate analyses for licensing purposes are reviewed: this is true for probabilistic and deterministic analyses and specifically for the determination of radiological consequences.

3) Structure of the BEPU: a) categorization of Postulated Initiating Events (PIE), b) grouping of events, c) identification of analysis purposes, d) identification of applicable acceptance criteria, e) setting up of the 'general scope' Evaluation Model (EM) and of related requirements starting from the identification of scenario related phenomena, f) selection of qualified computational tools including assumed initial and boundary conditions, g) characterization of assumptions for the Design Basis Spectrum, h) performing the analyses, i) adopting a suitable uncertainty method.

4) Under the item 3g): the roadmap pursued for the analysis foresaw the use of nominal conditions for the NPP parameters and the failure of the most influential system. The implementation of such roadmap implied the execution of preparatory code run per each scenario where all NPP systems were simulated. This also required the simulation the control and the limitations systems other than the protection systems. Once the 'nominal system performance in accident conditions (following each PIE)' was determined, it was possible to select the worst failures and calculate a new (i.e. the 'binding one') accident scenario.

5) Under the general scope of item 3e): several computer codes and about two dozen nodalizations have been used, developed and, in a number of cases, interconnected among each other.

6) Qualification was necessary for the computational tools mentioned under item 5), within the framework depicted under item 3). The issue constituted by qualification of code-nodalization user was dealt with in the same context. Specific methods or procedures including acceptability thresholds have been developed and adopted.

7) Under the scope of item 3i): the uncertainty method based on the extrapolation of accuracy, developed at University of Pisa since the end of 80's, was used to create the CIAU (Code with capability of Internal Assessment of Uncertainty) and directly used for quantifying the errors in the calculations, as needed.

The execution of the overall analysis and the evaluation of results in relation to about one-hundred PIE revealed the wide safety margins available for the concerned NPP that was designed in the 80's.

Risk Concept in Nuclear Industry

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The concept of risk evaluation is existing for centuries. It is present everywhere, in the economy, in the industry and in our everyday life. In the decades past, the role of risk concept is increasing continuously, which is due to a lot of reasonable causes. The paper shows, that why the risk-based applications are extended in more and more branches of industry. This likes to be a "communication interface" between the "economists" and "engineers". This new improvement in inpection planning involves numerous benefits, but raises some questions, e.g. from the regulatory point of view. What are needed to implement the risk-based strategy and for what can we expected as the result of it? The paper shows a solution of the implementation of the risk-based methodologies, which main principles can be applied in the most fields of industry.

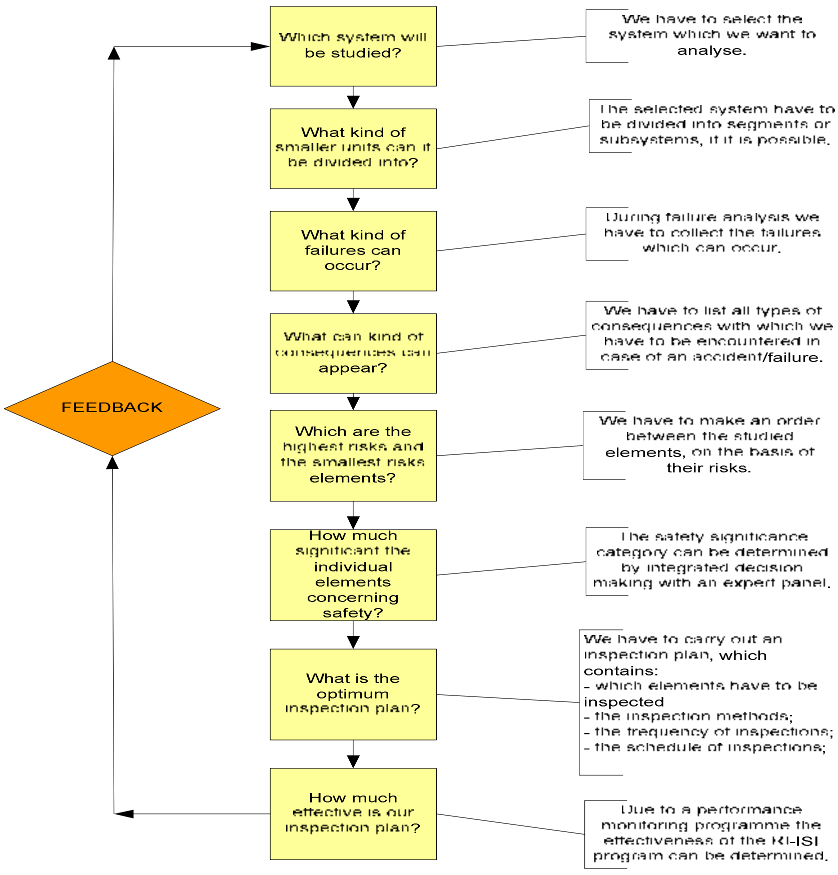

FIG. 1. Main steps of a risk-informed concept

The main aim is to focus the maintenance and inspection efforts on the highest risk elements, but how can we establish an adequate order between them, on the basis of risk? Are there any standards or guidelines for risk-based strategies in the different industries? The paper tries to summarize the current state of risk-based approaches, considering the benefits and difficulties of it. The paper concentrates on the nuclear industry, which is one of the most controversial industry, since the whole society is divided on this issue. The concept of risk is unthinkable in this field for a lot of people, including some regulatory bodies. As opposed to it, in some countries there are an acceptable process is the risk-based managing of nuclear power plants. What is the cause of this sharp difference? The purpose of this article is to demonstrate the state of the art in this field and to show the trends in the development of risk-based approaches

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Application of the Performance Validation Tool for the Evaluation of NSSS Control System Performance

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Nuclear steam supply system (NSSS) control systems play a vital role in operating the nuclear power plants (NPPs) within their designed condition. Therefore, lots of efforts have been made to screen out errors existing in these systems and check the performances when the control systems were either retrofitted in operating NPPs or installed in newly constructed NPPs because the remaining error would cause unexpected transient during normal operation and trip the reactor. However, it is difficult to filter out all errors in design process because some errors occur as a result of the combination of a dormant error and the onset of the specific set of conditions that triggers the error [1]. Therefore, many dynamic tests have been performed during power ascension test (PAT) for newly constructed NPPs to find out errors remained. However, performing actual dynamic tests increases risk and economical losses in case of control system retrofit in the NPPS in operations. Therefore, it is necessary to simulate these dynamic tests in this case.

A performance validation tool was developed to simulate the dynamic test. This tool was successfully applied to the hardware replacement project for Yonggwang (YGN) 3&4 feedwater control system (FWCS) in 2008. The FWCS hardware was directly connected with the windows based nuclear plant performance analyzer (Win-NPA) which is an interactive, high fidelity, and real-time engineering simulator. After preparing test equipment by replacing the FWCS model in the Win-NPA with a real FWCS hardware, dozens of performance related transients were simulated for the existing and new FWCS hardware. The results were compared to prove that the new FWCS hardware was properly manufactured in accordance with design specifications. Also, the performance of the new hardware was evaluated by comparing with that of the existing hardware [2].

On the basis of successful experience in YGN 3&4, this performance validation method using the Win-NPA extends its scope to newly constructed NPPs. If dynamic tests are simulated before the PAT with the Win-NPA connected to NSSS control systems hardware, errors in the control systems can be found and corrected before the PAT. Furthermore, the performance of control valves also can be tested and evaluated with the configuration shown in figure 1. It is widely known that about 30% of control problems are caused by valve problems such as stiction of positioner, low or large air flow, and improper size. As the performance of control systems and control valves is in advance confirmed before the PAT, unwanted transients or reactor trip due to control problems will be avoided during the PAT.

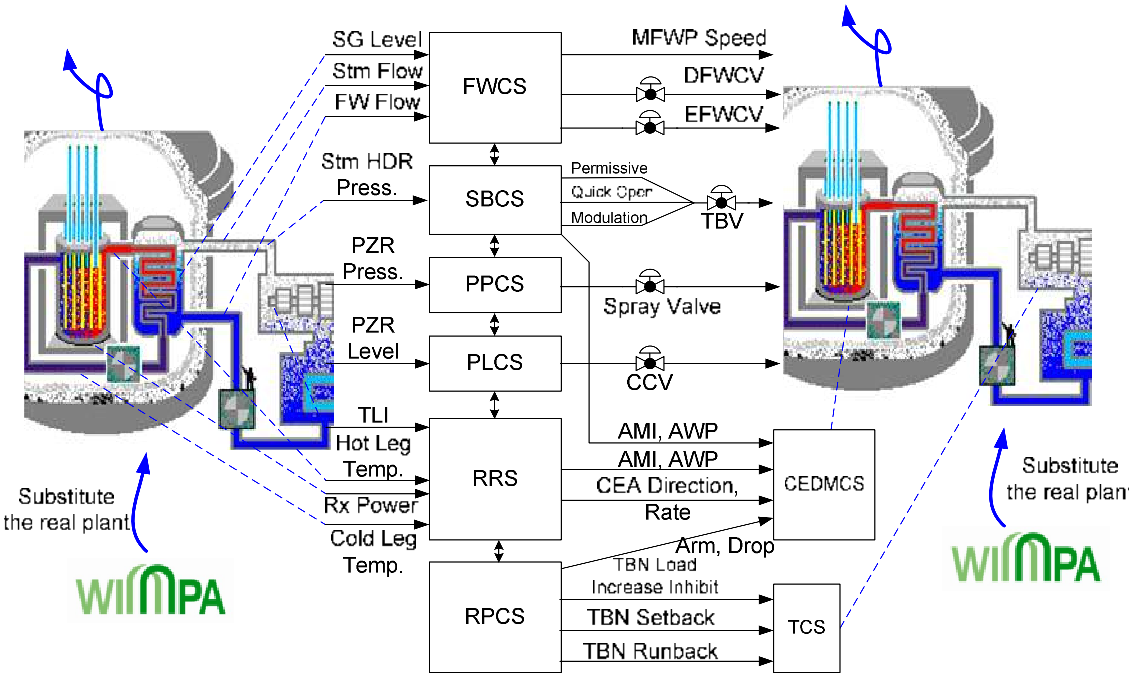

Fig. 1 Interfacal configuration between performance validation tool and control systems and control valves

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NULIFE ASSOCIATION: an Opportunity to Face Gen II / Gen III Challenges through Cooperative Research

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The European Network of Excellence NULIFE (Nuclear Plant Life Management) has a clear focus on integrating research for Long-Term Operation (LTO) to support Generation II plants safe and competitive operation, as well as to develop common basis and tools for current and evolutive Generation III nuclear power plants. NULIFE shall for instance help in the development of design criteria for future generations of nuclear power plant. NULIFE was kicked-off in October 2006 and will work over a 5-year period to create a single organization structure, named NULIFE ASSOCIATION, sustainable and capable of providing cooperative R&D at European and international level to the nuclear power industry. Joint research and development activity priorisation procedure and first R&D projects are defined to be started late 2009 or early 2010. Led by VTT (Technical Research Centre of Finland), the project has a total budget of around EUR 10 millions, with today 44 partners drawn from leading research institutions, technical support organisations, power companies and manufacturers throughout Europe. While over half are from the research sector, NULIFE also involves many industrial organizations and, in addition to their R&D contributions, these take part in a dedicated End User Group. The End User Group, led by EDF, consists today of almost all european nuclear utilities.

NULIFE has just completed :

- *A coherent working structure*
- *Links to National programmes and achieving good progress in the Expert Groups and Pilot Projects*
- *An active participation of End User Group in drafting the research strategy*
- *The basis for the business plan and an operational model to create soon NULIFE Association*
- *Joint participation with Euratom and Rosatom (CEG PLIM) and other international activities.*

The current business plan of NULIFE includes the establishment of an international association (NULIFE Association). NULIFE Association will be sustainable and shall take over the Network of Excellence in roughly two years. It shall act as a facilitator to cooperative research. It shall :

- Trigger innovation and promote new ideas
- Implement support to all Plant Life Management (PLIM) related issues.
- Support the launching of new R&D projects
- Find and link funding partners and find R&D capabilities from the NULIFE competence pool

- Support project consortium agreement and management
- Support integration and harmonisation.

Strategic research planning was started together with business plan and has been strongly related to the preparation of the First Chapter of the Strategic Research Agenda (SRA) of the Sustainable Nuclear Energy Technology Platform (SNE-TP). Many NULIFE organisations and members participate actively in the SNE-TP

- Governing Board: Chair, CEA, VTT, E.ON, EDF, Vattenfall, AREVA, SCK-CEN, TECNATOM, PSI, NRG,…
- Executive Committee: Chair VTT
- Strategic Research Agenda Group: Chair of subgroup Gen2/Gen3, EDF
- Deployment Strategy Group: EDF.

The First Chapter of the SNE-TP SRA defines the strategic targets in Long Term Operation (LTO), Performance improvement and External factors. In the Long Term Operation area, Safety justification, Ageing mechanisms of Systems-Structures-Components, Ageing monitoring and Prevention and mitigation of ageing are important subjects. In addition the SRA considers some cross-cutting areas like structural materials, prenormative research, codes and standards, modelling, simulation and methods.

NULIFE is seen as a key instrument in implementing PLIM related topics of the SNE-TP strategy. The on-going and near future actions are on one side the preparation of road maps and specific short, medium and long term research topics to deploy the SRA, and on the other side to facilitate the launching of R&D projects on needs already shared by several stakeholders. Three projects are under construction on civil work, thermo-hydraulic code benchmark and warm pre-stress effect. Each project is supported case by case by some members of the NULIFE End User Group, which is now very representative with around fifteen participating utilities, among which the 8 leading European nuclear utilities.

Model Reference Control & Protection Theory and Implementation for Nuclear Power Plants using Real-time Simulations

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This paper shows that when combining the real time data from nuclear power plant measurement equipment with a real time simulator of the plant it is possible to perform tighter control over the plant and unwanted plant operation can be recognised earlier. As soon as the unwanted operation is identified as a fault, the fault cause, severity and location in the plant can be deduced.

The traditional protection and control of nuclear plants by measuring variables inside the plant and initialising processes based on predefined rules set up from the design knowledge base - is effective but not without shortcomings. As plants become more complex the predefined protection and control rules are becoming more complex and a very large safety margin has to be designed into these rules [1]. Information from measurement equipment provides a picture of the plant at any time, but without predefined operational limits, it can not provide any indication of the expected behaviour of the plant.

The advancements in processor speeds and the development of sophisticated numerical algorithms allows the simulation of plant processes in real time [2]. The simulator, however, can only predict expected plant behaviour because it is blind to faults occuring in the plant.

Combining the information from the measurement equipment with the ability of the simulator to predict what should be happening inside the plant in real time provides a very effective method to address the shortcoming of the existing protection and control philosophies.

The model control & protection theory consists of two parts:

- A dynamic operating window around the operating point that moves along with the operating point, this allows tighter control as well as early reconition of unexpected behaviour.
- Fault isolation & identification, once unexpected behaviour has been recognised the fault data is isolated from expected transient information and analysed to classify the fault into a category, location, size and cause.

THE DYNAMIC OPERATING WINDOW

In theory the measured operating point of the nuclear plant should always be predicted exactly by the plant simulator and any variation from the predicted dynamic operating point constitues faulty behaviour. Taking into acount equipment and calculation inacuracies, the predicted operating point is enlarged to a predicted dynamic operating window around the measured operating point. Without the use of a simulator providing a small dynamic operating window the protection system utilises a large static operating window only based upon the safe operating constraints of the plant equipment. The position of the operating point inside this large operating window can not give any indication as to the correct operation of the plant, and the plant is deemed to operate correctly as long as the operating point remains inside the static operating window. The use of a dynamic operating window has two distinct advantages:

- During expected transients no fault would be flagged since the operating point will remain inside the operating window.
- During unexpected transients, the operating point will move outside the small operating window quickly and a fault condition would be flagged earlier.

FAULT ISOLATION AND IDENTIFICATION

We isolate the fault data by recording real measured data from the plant, and comparing it with expected real time data from the plant simulator. The difference between the two data sets constitute unexpected plant operation information. This is possible because the real-time simulator only provides information on the expected transient and no information on the fault. Noted that this calculation is not always a linear subtraction.

Combining the normalised fault information from various measurements inside the plant and correlating this information with various pre-simulated faults identifies the fault with an increasing level of certainty as the fault effects unfold over time and this information becomes available to the real time system. The following sequence is followed:

- The system identifies that the plant is behaving unexpectedly.
- The unexpected behaviour is isolated. The fault is categorised by matching the behaviour with known behaviour of certain fault categories.
- The fault is identified by following behaviour trend of a number of measured variables.
- The position of the fault is identified by triangulating the measurement equipment that indicated the fault first.
- The magnitude of the fault is calculated from rate of change in the time domain.

SUMMARY

Our research show that it is possible to generate a reference fingerprint for various faults based on its cause and position in the reactor. In these studies the normalised fault remains within 0.5% of the reference fault at all times. This enables plant diagnostics to identify a fault, the cause and the severity even if the plant is still be operating inside the traditional acceptable operating parameters during any transient. The system also improves dependability of the control & protection system of the plant by identifying faulty measurement equipment that would normally result in a plant trip.

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Technical Support to An Operating PWR vis-à-vis Safety Analysis

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Currently a PWR of 300 MWe capacity CHASNUPP-I is in operation since the year 2000. Technical support being provided includes In-core fuel management and corresponding safety analysis for the reshuffled core for the next cycle. Prior to start of cycle six an extension in cycle five based on coast down technique was achieved of almost 30 effective full power days.

Cycle 6 was designed to achieve the safe and economical loading pattern. The technique used is designated as out-in mode (modified). In this technique, most of the fresh fuel assemblies are not directly located at the periphery of the core, but near the boundary. This technique has the advantage that no burnable absorbers are used in each cycle and we get less radial neutron leakage and increased discharge burnup and cycle length. Operating experience/feedback shows that this type of loading pattern gives better economy without resorting to the conventional in-out technique. The lifetime of the cycle is predicted as 10371 MWD/MTU or 373 Effective Full Power Days (EFPD at 998.6 MWth). In design calculations, the end of cycle is reached at 10 ppm critical boron concentration in the unrodded core.

Measured critical boron concentration at HZP, BOL is 1453 ppm compared with the calculated value i.e 1457 ppm, is within the acceptable limits. It is also observed that the calculated reactivity worth of T1 is -1771 pcm as compared to measured value i.e -1802 pcm with difference of only 1.6 % showing the reliability of the design value. The measured Moderator temperature coefficient (MTC) is 2.52 pcm/^{0}C at all rods out (ARO) and critical boron concentration (CBC) condition whereas the calculated value is 3.36 pcm/^{0}C (at predicted CBC of 1457) having a good agreement with design value.

Safety evaluation of cycle 6 was carried out for the reshuffled core. All the probable accident scenarios based on initiating events as given in the FSAR were evaluated with respect to input parameters. For a specific event, the comparison of critical safety related core physics parameters between the reference case and current cycle was undertaken. For instance the scenario related to reduction in feedwater temperature or addition of excessive feedwater flow which can cause an increase in the heat transfer from the primary side to the secondary side in the steam generator and a decrease in RCS temperature was examined with respect to the input parameters. In this event, the most negative moderator temperature coefficient (MTC) and the least negative Doppler coefficient (DC) are the critical parameters. Maximum MTC and Minimum DC were calculated as -52.4 pcm/^{0}C & -2.32 pcm/^{0}C compared to reference value (FSAR) -59.0 pcm/^{0}C & -2.0 pcm/^{0}C respectively. The critical parameters of Cycle 6 are thus bounded by the reference analysis. Similarly, another event was considered of partial or complete loss of forced reactor coolant flow which could result in a rapid increase in the coolant temperature reduction in power because of the negative reactivity insertion. For a conservative analysis, the least negative moderator temperature coefficient and the most negative Doppler coefficient are the critical parameters. Minimum MTC and Maximum DC are calculated as -1.9 pcm/^{0}C & -3.04 pcm/^{0}C compared to reference value 0.0 pcm/^{0}C & -3.6 pcm/^{0}C respectively. The critical parameters of Cycle 6 are bounded by the reference analysis. Since all initiating events evaluated have relevant parameters within the limits, it is concluded that reactor will operate safely during cycle 6.

Synopses for Topic 5

ADVANCED TECHNOLOGY APPLICATION

CAREM Prototype Construction and Licensing Status

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After several years of development the CAREM Project reached such a maturity level that the argentinean government decided the construction of CAREM prototype.

CAREM is a CNEA (Comisión Nacional de Energía Atómica) project. This project consists on the development, design and construction of a small nuclear power plant. First, a prototype of an electrical output of about 27 MW, CAREM 25, will be constructed in order to validate the innovation of CAREM concept and then up-rated to commercial version.

CAREM 25 is an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the reactor and also contribute to a higher level of safety:

— Integrated primary cooling system.

— Primary cooling by natural circulation,

— Self-pressurised.

— Safety systems relying on passive features.

The primary system is integrated, that means the whole high energy primary system -core, steam generators and pressurising system- and the absorbers rods drive mechanisms are contained inside a single pressure vessel. The core has 61 fuel elements with hexagonal cross section and different U 235 low enrichment. Each fuel element contains 108 fuel rods, 18 guide thimbles, and an instrumentation thimble. Reactivity control during normal operation is achieved by means of movable control rods and burnable poison. Strongly negative temperature and density feedback coefficients are the consequence of no boron use for reactivity control during normal operation, allowing the control system to keep control of the reactor power through transients and load variations with minimum control rod motion. The 12 steam generators are of a ‘Mini Helical' vertical, 'once through' design. Coolant in the primary and secondary sides flow in counter-current with secondary flow coolant flowing upwards inside the tubes. The secondary system exits the steam generator with ample superheating.

Reactor coolant circulates by natural circulation. The driving force is the density difference between the hot and cold legs of the loop. The coolant flow is defined by the location of the steam generators -cold source- above the core -hot source-. The coolant acts also as moderator.

Self-pressurisation of the primaiy system is the result of natural trend towards the liquid-steam equilibrium. Due to self-pressurisation, bulk temperature at the core output corresponds to saturation temperature at the primary pressure. Conventional PWR's heaters and sprinkles are thus eliminated. The large dome volume contributes to the damping of pressure perturbations. To shutdown the nuclear chain reaction and to keep the reactor in a sub-critical state. CAREM NPP has two different and independent safety systems activated by the reactor protection system or by the operator. They are the first shutdown system -neutron absorbing elements- and the second shutdown system –borated water injection system-. Hydraulically Drive Control Rods avoid the use of mechanical shafts passing through the primary pressure boundary since the whole device is located inside the RPV. Their design is an important development in the CAREM concept in case of blackout, the reactor decay heat is transferred through the Residual Heat Removal System, to a pool located in the upper level of the containment. The pool water is vaporized and then conducted to the pressure suppression pool, where is condensed.

Technical and economical advantages are obtained with the CAREM design compared to the traditional design:

— Large Loss of Coolant Accideat (LOCA) has not to be handled by the safety systems due to the absence of large diameter piping associated to the primaiy system. The size of possible break in the primary is 38 mm.

— Innovative hydraulic Drive Control Rods avoid Rod Ejection Accident.

— Large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or severe 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 pressurizer results in lower costs, added safety, and advantages for maintenance and availability.

Several activities are ongoing with the purpose of obtaining the Construction License for CAREM Prototype. The Preliminary Safety Analysis Report is under development in order to be presented by the end of this year. Siting activities such as soil studies and environmental analysis are being performed.

The construction of a high pressure and high temperature rig for testing the innovative Hydraulic Control Rod Drive Mechanism will be finished this year.

In the fuel element area, both the fuel pellets and the fuel elements itself are under development. Uranium dioxide, burnable poison oxide and the appropriate equipment for pellet manufacturing will soon be available. Fuel element dummies that will be used to analyze mechanical integrity and test the behaviour under different flow conditions are under construction.

The use of robotics and the development of a plant simulator are considered.

Contracts and agreements are being taking with different argentinean stakeholders to perfonn detail engineering.

The procurement process of main components such as the RPV is being started with local suppliers.

Concept of a High Pressure Boiling Water Reactor, HP-BWR

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INTRODUCTION

Some four hundred Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR) have been in operation for several decades. The presented concept, the High Pressure Boiling Water Reactor (HP-BWR) makes use of the operating experiences. HP-BWR combines the advantages and leaves out the disadvantages of the traditional BWRs and PWRs by taking in consideration the experiences gained during their operation. The best parts of the two traditional reactor types are used and the troublesome components are left out. HP-BWR major benefits;

1 SAFETY IS IMPROVED

The control rods are gravity operated instead of be operated by an intricate hydraulic system. The gravity operated control rod system has served well in PWRs. The stems are introduced into the reactor vessel via the vessel head. The control rods themselves are in the form of a cross, as it is in the BWRs. This assures large space for the cross formed rods between the BWR type fuel boxes. Also the neutron measurement sounds are introduced via the reactor pressure vessel head the way it is used in BWRs. All the pipe connections to the reactor vessel are well above the top of the reactor core. This means that a major pipe break will not empty the reactor vessel. Therefore core spray is not needed. Internal circulation pumps are used to assure hydrodynamic stability. In this way the orifices at the fuel channel inlets are chosen so that the one phase pressure drop will dominate over the two phase pressure drop to avoid hydrodynamic oscillations. By utilizing natural circulation one could omit the circulation pumps. However the margin to avoid hydrodynamic oscillations may be reduced. Compared to the traditional BWR the HP-BWR has further advantages, namely improved thermal efficiency due to the higher temperature and further improved inherent stability due to the increased negative power reactivity coefficien,.calculated with the RELAP and PARCS codes. Transient calculations made with the MATLAB code proves the HP-BWR long term stability without the use of any control system. Using presently available PWR pressure vessel and presently available BWR fuel boxes the HP-BWR approximate power output would be some 1200 MWe.

2 ENVIRONMENT FRIENDLY

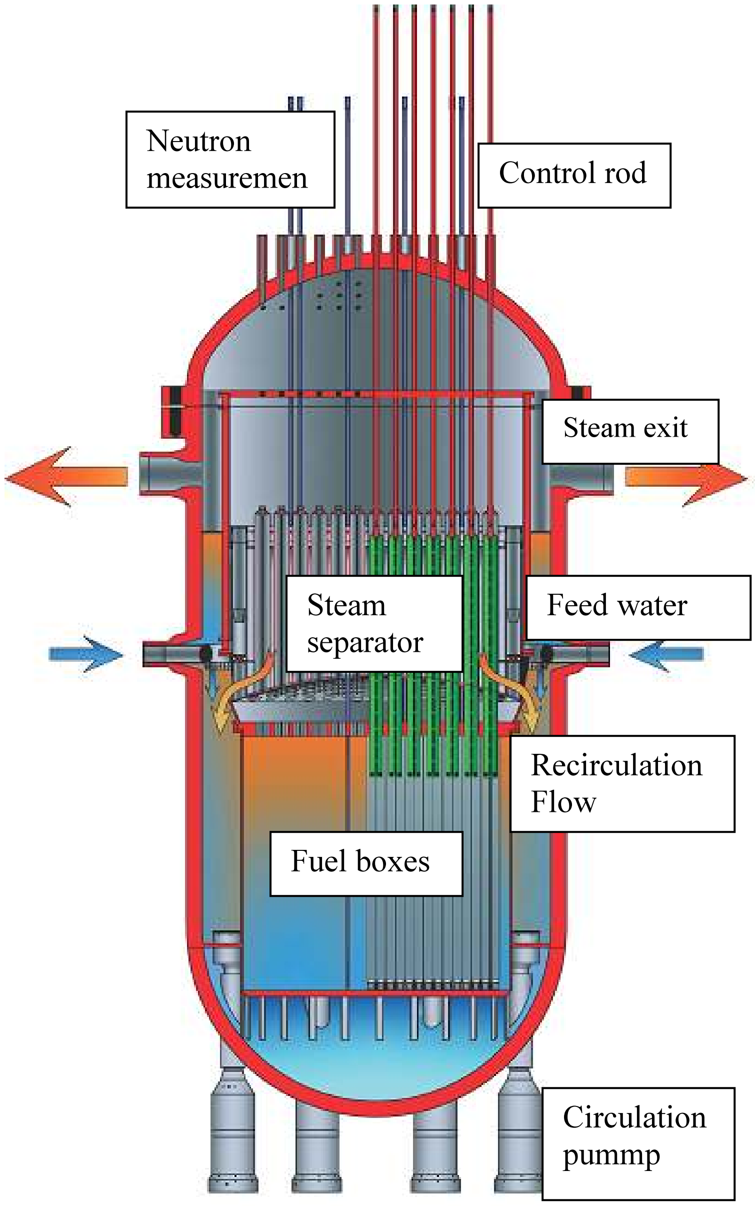
Improved thermal efficiency is attained by feeding the turbine with ~340^{0}C (15.5MPa) steam instead of ~286 ^{0}C (7MPa). The Carnot cycle theoretical efficiency (T_{Hot} - T_{Cold})/ T_{Hot} is for BWR ~46% and for HP-BWR ~51% at T_{Cold} = 300 ^{0}K, i.e. an increase by a factor of 1.109. Assuming the same improvement ratio, today's efficiency of ~33% would increase to ~37%. This demonstrates the advantage of the HP-BWR which utilizes the fuel more efficiently and releases less warm cooling water to the environment per produced kWh and consequently produces less waste. There are several conventional thermal power plants with 15.5 MPa turbines. Though to use dry saturated water might need some development work.

3 COST EFECTIVE SIMPLE

The HP-BWR operates in direct cycle mode, with no need for complicated and expensive PWR steam generators and also instead the perforated and rather complicated BWR reactor pressure vessel bottom a simplified smooth one is used. The main steam separators are inside the pressure vessel and secondary separators and dryers can be installed outside the reactor vessel, inside or outside the containment. The containment is a simple dry containment which allows easy entrance and inspections and also minor repairs during operation.

CONCLUDING REMARKS

As a reactor inspector on leave from Sweden I participated in IAEA's OSART and ASSET missions. Also due to my engagement at the International Electrotechnical Commission (IEC) I visited nuclear installations in Europe, Asia and America. This way I gained insight of the operational experiences of most reactor types. As a result, now I can contribute to the advancement of nuclear energy independently. As I have no obligation to any vendor or reactor type I can suggest an optimal reactor construction which hopefully will lead to further detailed studies at some vendors, power companies, research institutes and universities, especially after this conference. In Sweden already some universities expressed interest to make further studies of this concept.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SuperCritical Water-cooled Nuclear Reactors (SCWRs): Thermodynamic Cycle Options and Thermal Aspects of Pressure-Channel Design

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Research activities are currently conducted worldwide to develop Generation IV nuclear reactor concepts with the objective of improving thermal efficiency and increasing economic competitiveness of Generation IV Nuclear Power Plants (NPPs) compared to modern thermal power plants. The Super-Critical Water-cooled Reactor (SCWR) concept is one of the six Generation IV options chosen for further investigation and development in several countries, including Canada and Russia.

Water-cooled reactors operating at subcritical pressures (10 – 16 MPa) have provided a significant amount of electricity production for the past 50 years. However, the thermal efficiency of the current NPPs is not very high (30 - 35%). As such, more competitive designs, with higher thermal efficiencies, which will be close to that of modern thermal power plants (45 – 50%), need to be developed and implemented.

Previous studies have shown that direct cycles, with no-reheat and single-reheat configurations are the best choice for the SCWR concept. This paper presents two SCW NPP cycles based on direct, no-reheat and single-reheat regenerative concepts. The main parameters and performance in terms of thermal efficiency associated with these configurations is investigated in the first part of this paper. The cycles are generally comprised of: an SCWR, a SC turbine, one deaerator, ten feedwater heaters, and pumps. The SC turbine of the no-reheat cycle consists of one High-Pressure (HP) cylinder and two Low-Pressure (LP) cylinders. Alternatively, the SC turbine for the single-reheat cycle is comprised of one High-Pressure (HP) cylinder, one Intermediate-Pressure (IP) cylinder and two Low-Pressure (LP) cylinders.

There are a few technical challenges associated with the no-reheat and single-reheat SCW NPP configurations. The single-reheat cycle requires nuclear steam-reheat, thus increasing the complexity of the reactor core design. Conversely, the major technical challenge associated with a SC no-reheat turbine is the high moisture content in the LP turbine exhaust. A thermal-performance simulation reveals that the steam quality at the exhaust from the LP turbine is approximately 81%. However, the moisture can be reduced by implementation of contoured channels in the inner casing for draining water and moisture removal stages. The overall thermal efficiency of the two cycles was determined to be about 50% (assumptions are made to account for turbine and pump efficiency losses).

The SCWR core concept investigated in this paper is based on a generic pressure-tube reactor cooled with supercritical water. The considered reactor concept is based on a horizontal pressure-tube configuration with the following operating parameters: electrical power of 1200 MW, pressure of 25 MPa, reactor inlet temperature of 350°C, and reactor outlet temperature of 625°C.

In general, fuels currently investigated for the SCWR concept are high-temperature ceramics, similar to uranium dioxide (UO_2). Previous studies have shown that if UO_2 is used the centerline temperature of a fuel pellet might exceed the conservatively established industry accepted limit of 1850°C. As such, alternative fuel options, with higher thermal conductivities are being considered. The second part of this paper investigates a possibility of using uranium carbide (UC), uranium dicarbide (UC_2) and utanium nitride (UN) as SCWR fuels since they have higher thermal conductivities when compared to conventional nuclear fuels such as UO_2, MOX and thorium dioxide (ThO_2).

Also, important safety parameters such as a bulk-fluid temperature, heat transfer coefficient, inner sheath temperature and fuel centerline temperature have been calculated along the heated bundle-string length for non-uniform cosine-based Axial Heat Flux Profiles (AHFPs). To model a generic SCWR fuel channel, a 43-element bundle was assumed.

In addition, a new heat-transfer correlation for supercritical water flowing in vertical circular bare tubes was proposed. This correlation can be used for preliminary conservative estimation of heat transfer coefficients in supercritical water-cooled bundles as bundle correlations have not been developed yet.

Conceptual Design of Nuclear CHP Using Absorption Cycle

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This paper aims at providing a conceptual idea on the combined heat and power (CHP) using the absorption cycle to simultaneously generate both electricity and useful heat, which is applicable to the conventional nuclear power plants (NPPs).

The originality of the scheme is 1) it does not change the operation strategy of the NSSS, 2) the thermal energy of waste heat can be transferred to a long distance, and 3) the thermal energy can be used for cooling as well.

As it is expected that the number and the share of NPPs increases soon, the necessity of a partial load operation was raised in argument in case of South Korea. This means the surplus of nuclear energy. In order to make the best of nuclear fuels loaded once, we proposed a combined cycle instead of cutting back reactor power to meet a partial load demand. Figure 1 shows the schematic drawing of the proposal. Since a steam demand in the turbine cycle is equivalent even though an electricity demand is different, the operation strategy of the NSSS does not need to be changed. When a partial load demand is triggered off, turbine power is cut back and a bypass path is open. The bypass path is used for transferring waste heat to an absorption cycle.
The CHP using absorption principles was initially developed over 100 years ago. The absorption cycle is a process by which heating and/or cooling effect is produced through the use of two fluids and some quantity of heat input. The absorption cycles accomplish heat transferring through the evaporation of a refrigerant at a low pressure and the rejection of heat through the condensation of the refrigerant at a higher pressure. In the absorption cycles, a secondary fluid or absorbent is used to circulate the refrigerant. Absorption cycles are commercially available today in two basic configurations; lithium bromide/water and water/ammonia (respectively absorbent/ refrigerant) [1].

We can have several advantages in this idea. This principle can design a heat transfer mechamisn to convey thermal energy to a long range, which means the waste heat from nuclear stations can be used for practical purposes even in a populated district. District heating/cooling, industrial process heat supply, or seawater desalination are expected to be possible applications. In case of the absorption cycle, running cost is very low as long as waste heat is available. The nuclear CHP should be, therefore, one of the methods to maximize the efficiency of the entire cycle. There must be several disadvantages such as high capital cost, more space required, complex configuration, and so on [2].

The paper will provide simulation results for deciding thermo-dynamic viability and economic feasibility with comparing a few design alternatives [2].

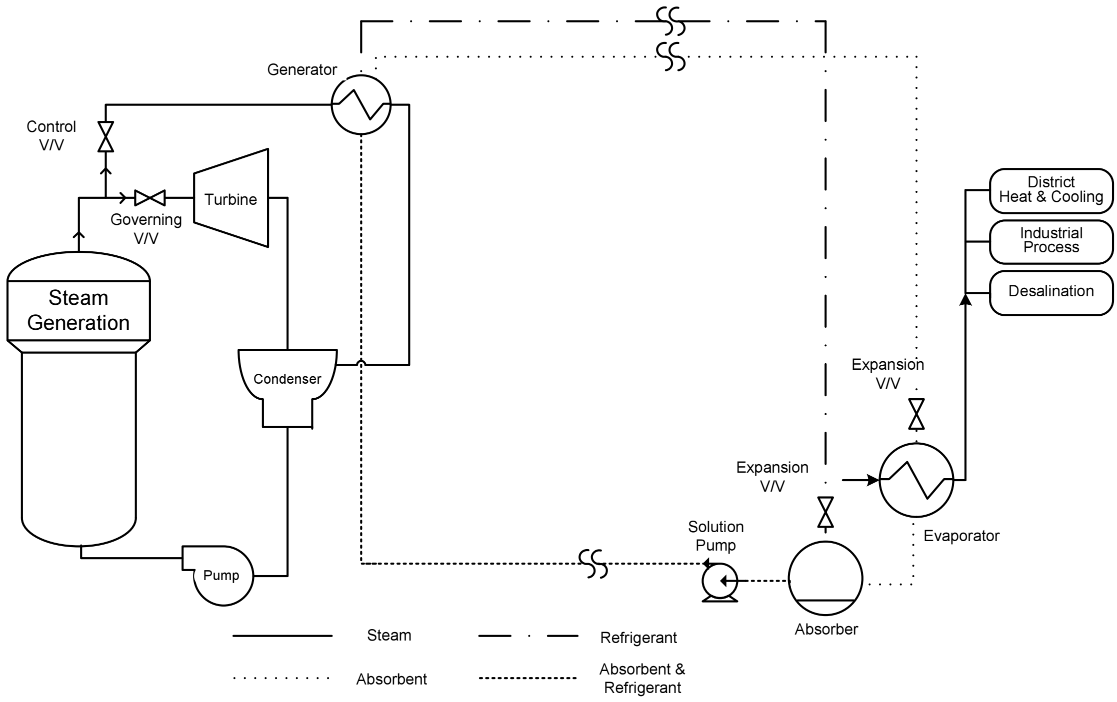

FIG. 1. Schematic drawing of the nuclear CHP cycle

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Water Cooled FBNR Nuclear Reactor

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The world with its increasing population and the desire for a more equitable and higher standard of living, is in the search for energy that is abundant and does not contribute to the problem of global warming. The answer to this is a new pardigm in nuclear energy; i.e., through the innovative nuclear reactors that meet the IAEA's INPRO philosophies and criteria that will guarantee the generation of safe and clean energy. The emerging countries to nuclear energy that are not in hurry for energy and look into the future are looking into the participation in the development of such innovative nuclear reactors. They can start developing the non-nuclear components of such reactors in parallel with creating the nuclear infra-structures according to the guidelines of the IAEA suggested in its milestones document. In this way, they can benefit from numerous advantages that the development of a high technology can bring to their countries be it scientific, technological, economic or political. A solution to the present world economic crisis is investing in such projects that contribute to the real econmy rather than speculative economy. This will help both local and world economy.

One such innovative nuclear reactor is the FBNR that is being developed with the support of the IAEA in its program of Small Reactors Without On-site Refuelling. It is a small (70 MWe) reactor with simple design based on the proven PWR technology (www.sefidvash.net/fbnr). The simplicity in design and the world wide existence of water reactor technology, makes it a near term project compared to other future reactors.

Small reactors are most adequate for both the developing and developed countries. They require low capital investment, and can be deployed gradually as energy demand calls for. The generation of energy at the local of consumption avoids high cost of energy transmission. The pardigm of economy of scale does not apply to the FBNR as it is a small reactor by its nature. The FBNR enjoys the economy of mass production. FBNR can serve a dual purpose plant generating electricity and producing desalinated water at the same time at lower cost.

The FBNR has been evaluated by the IAEA's INPRO Methodology from the safety and non-proliferation points of view and is shown to be a fool proof reactor against nuclear proliferation and have inherent safety against any concievable accident.

The reactor has in its upper part the reactor core and a steam generator and in its lower part the fuel chamber. The core consists of two concentric perforated zircaloy tubes of 31 cm and 171 cm in diameters, inside which, during the reactor operation, the spherical fuel elements are held together by the coolant flow in a fixed bed configuration, forming a suspended fixed core. The coolant flows vertically up into the inner perforated tube and then, passing horizontally through the fuel elements and the outer perforated tube, enters the outer shell where it flows up vertically to the steam generator. The reserve fuel chamber is a 60 cm diameter tube made of high neutron absorbing alloy, which is directly connected underneath the core tube. The fuel chamber consists of a helical 40 cm diameter tube flanged to the reserve fuel chamber that is sealed by the national and international authorities. A grid is provided at the lower part of the tube to hold the fuel elements within it. A steam generator of the shell-and-tube type is integrated in the upper part of the module. A control rod can slide inside the centre of the core for fine reactivity adjustments. The reactor is provided with a pressurizer system to keep the coolant at a constant pressure. The pump circulates the coolant inside the reactor moving it up through the fuel chamber, the core, and the steam generator. Thereafter, the coolant flows back down to the pump through the concentric annular passage. At a flow velocity called terminal velocity, the water coolant carries the 15 mm diameter spherical fuel elements from the fuel chamber up into the core. A fixed suspended core is formed in the reactor. In the shut down condition, the suspended core breaks down and the fuel elements leave the core and fall back into the fuel chamber by the force of gravity. The fuel elements are made of UO_2 micro spheres embedded in zirconium and cladded by zircaloy.

Any signal from any of the detectors, due to any initiating event, will cut-off power to the pump, causing the fuel elements to leave the core and fall back into the fuel chamber, where they remain in a highly subcritical and passively cooled conditions. The fuel chamber is cooled by natural convection transferring heat to the water in the tank housing the fuel chamber.

The nest step in the development of FBNR is the construction of its prototype. Efforts are being made to secure participants in such an endeavor.

Supercritical Water-Cooled Reactor (SCWR) Development through GIF Collaboration

SCWR System Steering Committee Members

GIF SCWR Steering Committee[1]

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The Generation IV International Forum (GIF) was established to conduct collaborative R&D that will lead to the development of the fourth generation of advanced reactor systems. Six reactor concepts were selected for further development through GIF collaboration: 1) Very-High-Temperature Reactor (VHTR), 2) Sodium-cooled Fast Reactor (SFR), 3) Super-Critical Water-cooled Reactor (SCWR), 4) Gas-cooled Fast Reactor (GFR), 5) Lead-cooled Fast Reactor (LFR), and 6) Molten Salt Reactor (MSR).

The Supercritical Water-Cooled Reactor[2] (SCWR) is the only GIF concept that uses water as coolant and is, therefore, a natural evolution of current advanced water-cooled reactor technologies. Furthermore, the SCWR incorporates advances from supercritical fossil power plant technologies that have been operating successfully for a number of years. The main fossil power plant technology that will be used in the SCWR is supercritical turbines that can be incorporated in a direct thermodynamic cycle to increase thermal efficiency. In addition, using a direct cycle at supercritical conditions simplifies the plant and eliminates certain components, which results in significant reduction in capital cost. Other experiences from fossil plant operation related to materials performance and water chemistry will be useful for the SCWR. This combination of advanced water-cooled reactor technology and advanced supercritical fossil technology is expected to result in a reactor concept that can be used to generate base-load electricity very economically and efficiently. This feature makes the SCWR a very attractive concept for utilities, especially those that have experience with both water-cooled reactors and supercritical fossil plants.

While the SCWR is highly rated in economics, it can also have significant improvements in other metrics such as safety, sustainability, and proliferation resistance and physical protection (PRPP). In the safety area, the starting point is to use safety features that are employed in current GenIII and GenIII+ technologies with the objective of providing further enhancements, wherever possible. In the sustainability area, the increase in thermal efficiency caused by the high outlet temperature (up to 625^{o}C at 25MPa) provides initial improvement in resource utilization. However, the SCWR can also be designed as a fast reactor (due to the significant decrease in the density of water above the pseudo-critical point), which provides opportunities to introduce advanced fuel cycles that aim at improving the sustainability and proliferation resistance metrics. In the area of physical protection, enhancements will be incorporated in the early design stages by incorporating lessons learnt from operating water-cooled reactors.

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[1] Membership from Canada (H. Khartabil; D. Brady), Euratom (T. Schulenberg), France (M. Delpech), Japan (Y. Oka; K. Yamada), and Rep. of Korea (Y. Bae)
[2] Operates above the thermodynamic point of water (> 374°C and 22.1 MPa)

Current SCWR R&D efforts clearly demonstrate the flexibility of this concept and the options that are available to design a practical and viable system. For example, the SCWR can be designed with a thermal, fast, or a mixed energy spectrum due to the significant change in water density as the temperature changes across the pseudo-critical point. In addition, the SCWR can be designed using the successfully deployed pressure-vessel (PV) or pressure-tube (PT) reactor technologies. These options have resulted in a number of conceptual designs within the GIF SCWR system that aim at providing options to provide significant improvements in all four GIF metrics: 1) Economics, 2) Safety, Sustainability, and 4) PRPP. The current SCWR conceptual designs under consideration by the GIF SCWR members include:

1. University of Tokyo thermal and fast spectrum designs: these are pressure-vessel designs known as the Super LWR (thermal version), and the Super Fast Reactor (fast version). Another thermal version design is being developed through collaboration between academic, research, and private organizations.

2. High Performance Light Water Reactor (HPLWR) under development in Europe.

3. CANDU®[3] -SCWR: this is a pressure-tube reactor that is being developed by AECL.

4. SCWR-SM: this is a pressure vessel design under development in the Republic of Korea.

5. Mixed core design (SCWR-M): this is a pressure vessel concept that is being developed at Shanghai Jiao Tong University.

The above designs have very similar challenges in combining existing advanced water-cooled reactor technologies with supercritical fossil technologies. This provides opportunities for collaboration in common R&D areas that are needed for any of the SCWR conceptual designs. For example, challenges in the selection of materials for the core components and the specification of suitable water chemistry require significant R&D that is common to all designs. In addition, R&D will be needed in other areas such as thermal-hydraulics and safety to extend the range of existing data to supercritical conditions. New or upgraded facilities that operate at supercritical conditions will be needed to perform in-core and out-core tests to produce data and information needed for design and licensing. A System Research Plan (SRP) for the SCWR has been developed by the GIF SCWR System Steering Committee (SSC) that outlines the R&D requirements for the SCWR development. The GIF SCWR members that are currently active in the SCWR R&D include: Canada, Euratom, France, Japan, Republic of Korea, and China (observer).

This paper presents the major features of the SCWR conceptual designs under consideration by the GIF SCWR members as well as relevant ongoing and planned R&D efforts.

[3] CANDU – Canada Deuterium Uranium, a registered trademark of Atomic Energy of Canada Limited (AECL).

Nuclear Co-generation Desalination Complex with Simplified Boiling Water Reactor VK-300

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With regard for the global-scale development of desalination technologies and the stable growth demand for them, Russia also takes an active part in the development of these technologies. Two major aspects play a special role here: they are providing the desalination process with power and introducing new materials capable to make the production of fresh water cheaper and raise the technical reliability of desalination units.

. In achieving these tasks, the focus is on the most knowledge-intensive issues, to which Russia is capable to make its contribution based both on the experience of developing national nuclear power and the experience of developing, manufacturing and operating desalination units, including the use of nuclear power (the experience of BN-350 in Aktau (formerly Shevchenko), Kazakhstan.

In terms of design, Nuclear Desalination Complex (NDC) with VK-300 reactor facility is a modification of a nuclear power unit with VK-300 reactor developed for application at Russian nuclear cogeneration plants. A power unit with VK-300 reactor has a design power of 250 MW(e) with the turbine unit operation in the condensation mode. In modes with the heat supply for desalination needs, up to 400 Gcal/h of thermal energy can be used as a steam from turbine extractions with the simultaneous electricity generation by the turbine generator of about 150 MW.

The report considers a VK-300 reactor based NDC with MED based distillation desalination units with horizontal-tube film evaporators. Russia has an extensive experience of commissioning and long-term commercial operation of domestically built desalination units with horizontal-tube evaporators of different power (from 0.1 to 700m^3/h). Seawater desalination units built on their basis are more economic than evaporators of other types - by the factor of 1.5-2.0 in terms of the energy consumption and by the factor of 1.5-1.8 in terms of the specific quantity of metal and the development area. With regard for the power unit capabilities of supplying heat for desalination (200-400 Gcal/h) as part of an NDC with a VK-300, it is expedient to use distillation units with a higher unit capacity.

The most attractive option is coupling of the VK-300 energy source with distillation desalination units operating based on the multi-stage evaporation principle (MED). This is the effective NDC structure allowing the use of turbine steam extractions for heat supply (via the intermediate circuit) to the desalination system producing high-quality distillate. As it provides with thermal energy a desalination complex with the capacity of 300.000 m^3/day, a nuclear plant consisting of two VK-300 power units allows production of distillate with the cost of 0.58 dollars/m^3. In this case, the electricity supply to the power system is 357 MW(e). The electricity cost is 0.029 dollars/kWh.

Plutonium Management in Small Nuclear Country

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Quallified guess of SNF Pu content in Slovakia can be performed as recapitulation of Pu amount in spent fuel of already stopped NPP V1 Bohunice in Slovakia was performed [1]. Spent fuel of 2 units VVER-440, operated in altogether 55 core cycles (1-year) encompasses 5143 FAs. Spent FAs from first cycles, moved to Soviet Union, are not included. Masses of important nuclides - see Tab.1. It can be seen, that about 5.6 t of Pu is contained in the NPP V1 spent fuel.

Nuclide	Mass [kg]
Sum of uranium	5.90547E+05
Sum of neptunium	3.33015E+02
Sum of plutonium	5.62261E+03
Sum of americium	8.51769E+02
Sum of curium	2.78415E+00
Sum of californium	1.02119E-13
Sum of actinides	5.97357E+05
Sum of fission products	1.54463E+03
Total	5.98902E+05

Tab.1 Element masses in the spent fuel of NPP V1 Bohunice

Taking into account, that other VVER-440 units in Slovakia (V2 - 2 units in Bohunice, 2 units in Mochovce) have finished up to now at least 65 1-year core cycles, we can count more than 6 t of Pu in it's SF and we can suppose, that at least 11 t of Pu in SF is available in Slovakia.

Fast reactor SUPER PHENIX can serve as an example of FR. Mass of its core part with Pu enrichment 23.2 % is about 37.1 t of HM. It means, that for the start-up FR needs about 8.6 t of Pu [2].

It means, that even if FR (SUPER PHENIX type) is started in Slovakia, surplus of Pu is nowadays at least 2 t Pu and this value is growing instantly, as 4 VVER-440 units are in operation and hopefully fleet of LWR reactors in Slovakia will be extended in the future.

It seems to be reasonable not to increase or at least to limit Pu mass growth. In the Slovakia situation it can be performed with existing type of reactors - VVER-440. Without radical unit reconstructions it is possible to evolve fuel cycle in the directions: classical MOX, inert matrix fuel or thorium fuel cycle.

Classical MOX do not limit significantly Pu production in comparison with OFC. Three partially closed fuel cycles were compared numerically [3]. Reprocessing of UOX spent fuel is taken into account in all of them. In the cycle with zirconium inert matrix (IMF) Pu and MA from UOX reprocessing are reused in the self-cleaning manner. In the case of thorium cycles Pu only (PuThOX) or Pu from UOX and U233 from ThOX reprocessing (UPuThOX) are reused. Open fuel cycle with classical UOX fuel (UOX) is used as reference.

Results of analysis by spectral code HELIOS can be seen in Tab.2. It can be seen, that all three partially closed fuel cycles have smaller Pu and MA generation rate in comparison with open UOX cycle and perform partial Pu transmutation. The smallest Pu and MA generation and the highest Pu transmutation rate - more than 70% - offers partially closed fuel cycle with IMF.

	UOX	**PuThOX**	**UPuThOX**	**IMF**
Pu initial (kg/tHM)	0	97.38	41.42	14.05
Pu in spent fuel after 5y cooling (kg/tHM)	12.37	51.39	15.25	3.21
MA in spent fuel after 5y cooling (kg/tHM)	1.48	6.15	3.39	1.22
Pu transmutation rate (%)	0	47.22	63.18	77.08
Pu transmutation rate (kg/TWhe)	0	13.22	17.64	7.32
Pu generation rate (kg/TWhe)	32.2	15.5	10.5	6.2
MA genetration rate (kg/TWhe)	3.8	3.3	2.9	2.7

Tab.2 Potencial of Pu transmutation

Mass of Pu in VVER-440 SNF produced up to now in Slovakia is sufficient for introduction of FR. Significant reduction of continuing Pu production can be reached without expensive and time-consuming unit reconstructions by introduction of closed fuel cycle with IMF or Th. This reduction can simplify construction of deep repository and solution of nonproliferation problems. Remaining Pu and MA can be transmuted later in future FRs or MSRs.

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Roadmap Design for Thorium-Uranium Breeding Recycle in PWR

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Thorium is about three times more abundant in nature compared to uranium and exists mainly as fertile isotope ^{232}Th. It is well-known that ^{232}Th can be converted to man-made fissile isotope ^{233}U with higher conversion ratio or even breeding in thermal reactor while the fuel lattice or the core is well designed.

Nevertherless, if the conversion of ^{232}Th to ^{233}U is only driven by other fissile isotopes (so-called "seeds" , such as ^{235}U or ^{239}Pu), it will be difficult to maintain a sustainable fuel recycle; and if there is any redesign on the fuel lattice or the core structure, the scheme shall become quite difficult for implementation. Therefore, the most attractive case is to achieve sustainable Throuim-Uranium (^{232}Th-^{233}U) Breeding Cycle in exsisting thermal reactor without any change to the design of fuel lattice or core internals.

The paper is focusing on designing a roadmap to finally approach sustainable Throuim-Uranium (^{232}Th-^{233}U) Breeding Cycle in current PWR, without any change to the fuel lattice and the core internals, but substituting the UOX pellete with thorium-based pellete.

At first, the paper will reveal that the ratio of conversion or breeding of ^{232}Th to ^{233}U is mainly dominated by eipthermal neutron and fast neutron, even though the bigger η factor (effective fission neutrons) of ^{233}U in thermal spectrum is benificial to maintain more neutron population so that more thorium can be converted to ^{233}U. As conclusions, the hardened spectrum in PWR will be benificial to improve the conversion efficiency from ^{232}Th to ^{233}U, and the thorium-based MOX fuel can have a better ratio of convertion or breeding than discrete thorium rod or assembly in moderator region of PWR.

Secondly, the paper will reveal that there is a threshold value for the ratio of initial ^{233}U inventory to ^{232}Th inventory. If and only if the ratio is less than this threshold value, the ^{232}Th-^{233}U MOX fuel could achieve breeding. For typical PWR fuel lattice, the thredhold value is around 0.02.

Series of calculations have been summarized in the paper for thorium-based MOX fuels blended with various fissile seeds, such as MEU, reactor-grade plutonium and reactor-grade ^{233}U. The calculation results have verified the conclusions deduced from the neutronics theorey, and demonstrated that it is possible to achieve sustainable Throuim-Uranium (^{232}Th-^{233}U) Breeding Cycle without any redesign for current PWR (see FIG. 1).

Based on the analysis to above calculation, the paper presents a roadmap to approach sustainable Throuim-Uranium (^{232}Th-^{233}U) Breeding Cycle in current PWR (see FIG. 2), which is composed of 2 stages. In the first stage, the recycled reactor-grade plutonium from current PWR spent fuel is used as seeds to mix with thorium; then, the thorium-plutonium fuels are loaded into PWR core and will produce higher purity reactor-grade ^{233}U; In the second stage, after the thorium-plutonium fuels are discharged from PWR core and stay in spent fuel pool for proper time, the fuels will be reprocessed, the reactor-grade ^{233}U will be extracted and used as seeds to mix with thorium; then thorium-uranium fuels shall be loaded into the periphery of PWR core to compose so-called "blanket" for Low-Leakage and long-cycle reload core design, in which Throuim-Uranium (^{232}Th-^{233}U) Breeding Cycle will be achieved. After the thorium-uranium fuels are discharged from PWR core and cooled for enough time, the fuel will be reprocessed and the breeded reactor-grade ^{233}U will be extracted, and then, the next round recycle could be started with extra reactor-grade ^{233}U accumulated.

The detailed PWR core design and evaluation with Throuim-Uranium (^{232}Th-^{233}U) blanket are not involved in this paper and will be presented in future report.

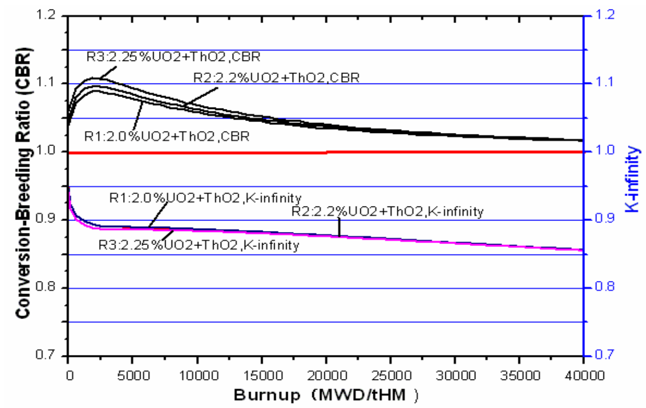

FIG.1. Conversion-Breeding Ratio and K-infinity vs. Burnup for Multi-Recycled Thorium-Uranium Fuel

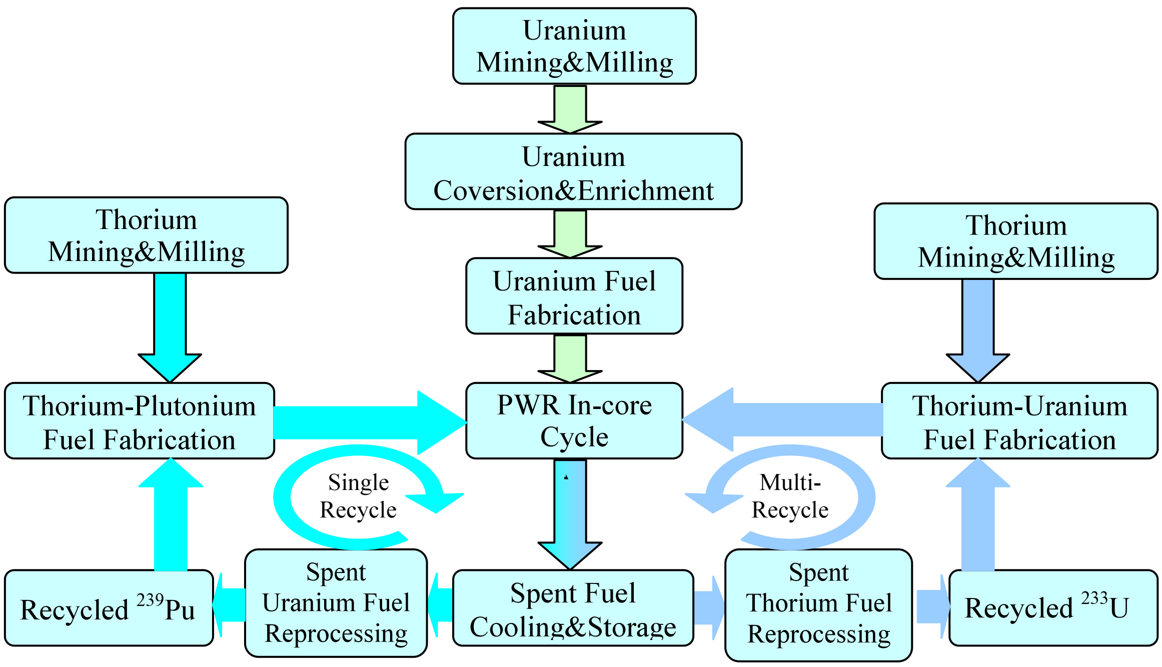

FIG. 2.Roadmap for Thorium-Uranium Recycle in PWR

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Rod-Type Quench Performance of Nanofluids Towards Developments of Advanced PWR Nanofluids-Engineered Safety Features

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Nanofluids, colloidal dispersions of nanoparticles in a base fluid such as water can afford very significant Critical Heat Flux (CHF) enhancement. Such engineered fluids potentially could be employed as advanced coolants in nuclear-engineered safety systems such as Emergency Core Cooling System (ECCS) and External Reactor Vessel Cooling System (ERVCS) with significant safety and economic advantages. When the potential application of nanofluids comes to ECCS, the situation of interest is quench phenomena of fuel rods during reflooding of emergency coolants. Therefore, we experimentally investigate the effect of nanoparticles on the cooling performance of the Inconel 600 cylindrical rod during quenching.

We have selected nanofluids with SiC nanoparticles for their high thermal conductivity and growing interests for nuclear applications. The nanoparticles are dispersed into de-ionized water to prepare a solution of 0.001 % by volume. In terms of colloidal stability, isoelectric point (IEP) is an important factor to decide whether colloidal particles can be stably dispersed in a base fluid. SiC nanoparticles have the known value of pH 2.5. The current nanofluid has a pH value far from the IEP which means the colloidal dispersion is stable without significant precipitation.

Nanoparticles have a type of spherical shape and 142 nm diameter in average with a normal distribution. It was checked that there were no significant changes of the physical properties through thermal-fluid characterization by using pH/conductivity meter, a transient heated needle method for thermal conductivity, viscometer, and a ring-type surface tension analyzer.

Quench phenomena have been investigated by using the nanofluid quench facility which consists of a data-acquisition system, a furnace, a cylinder tank for fluids and thermocouple-connecting rod-coupling cylindrical specimens. Cylindrical specimens were machined from Inconel 600; they were 12.5 mm in diameter and 60 mm long. The surface of all the specimens was polished to ensure or help repeatability of the quenching results. The specimens were equipped with type K sheathed thermocouple of 1.5 mm diameter buried at mid-length at a depth of 30 mm. The test procedure is first to preheat the quench specimen in the furnace until a predefined temperature is reached in the range of 800 ^{o}C-900 ^{o}C. Each specimen was quickly removed from the furnace to be quenched in 30 ^{o}C water. The temperature history was recorded on a personal computer at a frequency of 10 Hz. In addition, quenching phenomena are carefully investigated with initiation of quenching by using high-speed camera with 1024x1024 resolution image sensor and 3000 fps frame rate at full resolution. FIG. 1 illustrates centerline temperature histories recorded during quenching for water and SiC nanofluids, as well as atmospheric environment with the details in TABLE I.

TABLE I. Measured quenching time to drop to a temperature and cooling rates (CR)

	water	SiC-water	air
Max. CR	218.05	230.01	4.04
Temp. at Max. CR	585.05	613.58	833.34
Time to 600 °C	2.91	1.79	-
Time to 400 °C	3.98	2.84	-
Time to 200 °C	6.7	5.52	-

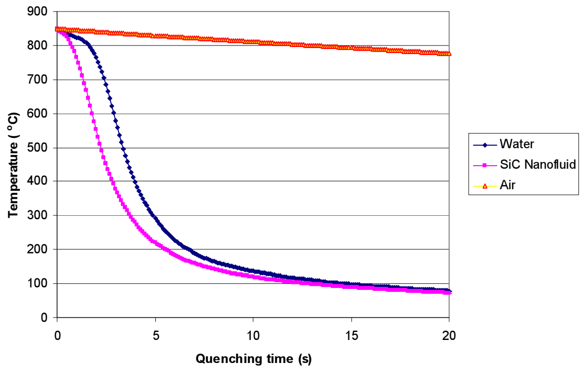

FIG. 1. Temperature history for the quench tests.

The present work experimentally investigated the effect of nanoparticles on the cooling performance of the Inconel 600 cylindrical rod during quenching. This paper provides the first insight to a rod-type quenching performance and phenomena of nanofluids with FIG. 2. Mechanistic changes expected from using nanofluid as a new coolant of an ECCS can be suggested as follows; (a) improved heat transfer coefficient of nanofluids in QF, (b) improved thermal dissipation accelerating QF, (c) locally nonuniform cooling in nanofluids, (d) rupture of vapor blanket/film due to turbulence enhancement, (e) improved radiation heat transfer of nanofluids, (f) improved surface wettability by nanoparticles. It is noted that the more detailed investigation should be done because of general trend of a nanotechnology lacking consistency. Therefore, a more systematic study of the effect of fluid temperature, nanomaterials and concentration on the quenching efficiency is underway in the lab.

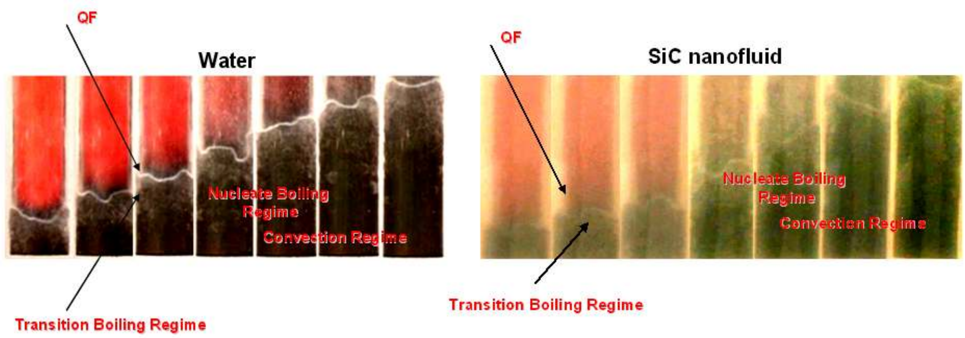

FIG. 2. Propagation of Quench Front (QF)

Application of Nuclear Energy to Oil Sands and Hydrogen Production

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Many novel and needed applications of nuclear energy arise in today's energy hungry, economically challenged world, and in solving tomorrow's search for a globally carbon constrained and sustainable energy supply. Not only can nuclear power produce low cost electricity, it can provide co-generation of process heat, desalinated water, and hydrogen with negligible greenhouse gas emissions. In each of these new applications, nuclear energy is competing against, or displacing conventional and established use of natural gas or coal in thermal power plants and boilers. Therefore, there must be a compelling case, in terms of supply certainty, stability, safety. security, and acceptability. In additon, a synergistic relation must exist or be created with the existing power and energy markets, the use of windpower, and the needs for low cost supply with negligible greenhouse gas emissions and carbon "footprint".

The development of Canada's oil sands resource depends on a substantial energy input for extraction and upgrading. So far, this input has been supplied by natural gas, a resource that (a) is a premium fuel; (b) has limited availability; and (c) produces significant CO_2 emissions. For the oil sands extraction process, natural gas is the current energy source used to generate the steam for in-situ heating, the power to drive the separation equipment, and the hydrogen for nominal upgrading before piping. However, there is a high demand and limited supply of natural gas, which results in projected higher prices and supply, and in future price and availability uncertainty.

We examine the applications of nuclear energy to oil sands production, and the concomitant hygrogen production, utilizing realistic reactor designs, modern power and energy market considerations, and environmental constraints on waste and emissions. We cover all aspects of feasibility, specifically technical issues, comparative economics, schedule, regulatory requirements, and other implementation factors. We compare and contrast the claims versus the realities, and also provide the synergistive utilization of co-generation of hydrogen using coupled nuclear and windpower.

Water Chemistry and Corrosion Prevention at Nuclear Plants with Supercritical Water Reactors

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The paper describes the main approaches and philosophy of water chemistry management at the new-generation plants with supercritical water reactors. The recommendations concerning the water chemistry management are based on the operating experience of both supercritical fossil plants and pressure-tube boiling water reactors AMB-1 and AMB-2 of the Beloyarsk NPP which used the nuclear steam reheated to 510 °C.

The paper discusses corrosion behaviour of structural rnaterials in supercritical water environment and suggests the ways of preventing pipeline and component corrosion in the various periods of plant operation.

Optimization of CARA Fuel Element with Negative Coolant Void Coefficient

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Analysis is made on CARA[1] nuclear reactor fuel element design to achieve a negative coolant void coefficient preserving its advantages on cost and operational performance.

The CARA fuel element is designed to replace the original fuel elements of both heavy water cooled NPPs in Argentina. The original advantages of the CARA fuel are its lower cycle cost compared with Embalse and Atucha I fuels and its lower linear power, which leads to greater DNBR and less pellet-cladding interaction (due to lower centre temperature in the fuel pellets). The key differences between CARA fuel and the original fuels are its larger number of rods (52 instead of 37) with collapsible cladding, its optimized enrichment in the sake of lower cost, its lower pressure drop design of spacer grid and its overall dimensions that allow it to be used in both NPPs.

In this analysis burnable poisons in the central zone of the fuel element are used to achieve negative coolant void coefficient, as suggested by A. R. Dastur et al.[2]. Different ^{235}U enrichments are also used in the rings of rods to compensate the effect of the neutron absorbers and preserve the neutronic economy. The CARA design is analyzed with multidisciplinary criteria: neutronics, economics and safety margins are taken into account.

Neutronic calculations are carried out with WIMSD-5 cell code. With the help of a code developed for this purpose several runs are made varying fuel parameters: amount, location and isotope of burnable poisons and enrichments in the four rings of fuel rods. The coolant void coefficient is calculated for a 100% void fraction and its dependence with burnup is observed to satisfy the requirements on the void coefficient over the entire in-core life of the fuel.

Economic evaluations are made with each fuel parameter variation to conserve the cost advantage of CARA over the original fuel elements. The costs compared include enriched uranium, burnable poisons, cladding and assembling for every new fuel element following the refuelling strategy of the original cores. First core cost is not taken into account as CARA is meant to replace an operating core and not to start a new NPP. Fuel cost is nivelated with a 15% discount rate. Enrichment cost is calculated in the base of Separative Work Units (SWU) price.

Power peaking factors in the fuel bundle are also studied for each fuel configuration as an operational restriction. Margins such as DNBR and linear power limit are tighter in few fuel rods on certain core locations. These locations have higher power due to core and fuel power peaking factors (PPF). Applying limits to PPF, DNBR and linear power remain on safe operating conditions as the core power does not change (CARA is a replacement fuel). Due to the higher number of rods in the fuel element a greater PPF is admitted and the use of higher enrichment rods in the outer rings that the greater PPF allows lowers the cost.

In the optimization process three different burnable poisons are tested, Dysprosium, Bore and Gadolinium, within the UO_2 rod or in separate rods of pure absorber.

Several runs are made evaluating the quantities mentioned above and optimization leads to an opposition among the merit figures. Establishing a required coolant void coefficient allows to select the best combination of cost – PPF, and hence obtain an optimum design of fuel element that complies with the design criteria.

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Synopses for Topic 6

SAFETY ASSESSMENT IN NPPS

Development of Universal Methodology of Specimen Free Nondestructive Inspection (Control) of Mechanical Properties of NPP Equipment Metal in All Stages of Lifetime

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Ever-increasing requirements for reliability and safety of equipment in nuclear power plants (NPP) dictate a necessity to obtain reliable and validated information about the condition of materials in the most safety-relevant and economically vital systems structures and components (SCC). Thus it is a state of science and technology approach to use one method, one methodic and one methodology to facilitate these goals with the purpose of keeping NPPs operating safely by virtue of knowing the state of ageing they are in (with respect to design limits and margins).

Method of the control/measurement/testing - how to conduct measurements; methodic - how to interpret the results of measurement; methodology - the program of the control/inspection and testing programmes: localities to conduct the tests, how often, and to follow evolution of test results with the aim of acting before a failure occurs.

Such methodology should be based on the use of specimen-free nondestructive method of the inspection (control), which could be used successfully at all stages of life cycle of the equipment: manufacturing, construction, installation of NPP, operation and during the NPP operation through integration into the Plant Life Management (PLiM) programme.

It will facilitate a real picture of change (degradation) of a SSC material's condition in the zones subjected to the harschest stressors (neutron irradiation, erosion-corrosion/flow, thermal fatigue, vibration etc).

Currently, there are various approaches used in the world to follow NPP ageing degradation, but until now, no specific methodology is used that could supply all the necessary information. Therefore, there is no way to use various results. Thanks to considerable advances over the last 20 years or so, the science of hardness testing offers an elegant, non-destructive way to obtain vital materials properties – even in-situ on operating SSC. In particular, the material's elastic-plastic condition may be measured, giving indications on tensile yield stress elevation due to hardening and also loss in ductility. The work-hardening index may be easily obtained, giving information on the ability of the material (e.g. pressure vessel steel and weld) to deform plastically without brittle fracture. However, the main lack of traditional methods of hardness measurement is absence of unification in carrying out of tests.

Taking into account the experience of the Center of Material Science and Lifetime Management Ltd. (CMSLM Ltd.) in the use of methods of hardness testing for the inspection of the equipment of NPP of Russian manufacture in Russia, Germany, Czech, Slovakia, Bulgaria, and also similar successful works in this direction in USA (Oakridge), to Czech (NRI Rez) and other countries, it can be seen that the most promising direction in the field of specimen-free inspection of mechanical properties by use of hardness and hardness-related characteristics is use of the kinetic indentation method (KIM, ABIT). This method is based on recording the process of elastoplastic deformation caused by the indentation of a ball indenter. This method allows one to obtain, besides hardness values, tensile properties, elongation, work hardening coefficient, true-stress/true strain diagrammes which normally required the destructive testing of small specimens. However, till now there is no universal method of interpreting the information obtained, although it is generally known that irradiation causes a loss in ductility and increase in hardness and lowering of the work-hardening coefficient. As base for development of such technique it is possible to use the new international standard (International Standard ISO/DIS 14577-1:2000-04 «Metallic materials – Instrumented indentation test for hardness and materials parameter», а также документ ISO/DTR 2938 «Metallic materials – Measurement of mechanical properties by instrumented indentation test – Indentation tensile properties»).

Thus, it is necessary to develop a uniform methodology of using KIM, ABIT with reference to the inspection of materials which will allow to unify the inspection of materials of various classes of the equipment over all stages of NPP life cycle. With the purpose of introduction of such a methodology it is necessary to develop and realize the program in the frames of IAEA with the above name. In the report are presented stages of this work.

Comparison Between International and Slovak Design Safety Requirements on Severe Accidents and Feasibility Analysis of the Safety Enhancement of VVER 440/V213 Plants to Comply with New Safety Requirements

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In accordance with the Defence- In-Depth philosophy, severe accidents have to be considered in the nuclear power plant (NPP) design independently of any measures implemented for their prevention. Consideration of severe accidents currently represents one of the development trends for new NPPs as well as for the safety upgrade of existing plants. This trend is in accordance with the revised set of the IAEA Safety Standards, in particular safety requirements for NPP design [1], and safety guides for design of reactor containment systems [2] and severe accident management programmes [3]. Special attention is also paid to Western European Nuclear Regulators' Association (WENRA) Reference Levels [4], developed on a consensual basis of the involved regulatory bodies and meant to be the minimum safety requirements for the existing NPPs.

Currently, new nuclear plants are also being designed and constructed with increased performance and a higher safety level, in accordance with utilities' expectations, such as expressed in European Utility Requirements (EUR) [5]. Due to the general practice in the nuclear industry to implement, as far as reasonable, new safety requirements into the design of operating plants, it is advisable to consult the currently-valid documentation of existing plants as well. This is in particular important due to another general trend, i.e., to ensure long-term operation of existing plant significantly beyond the originally-envisaged lifetime of the plants.

Slovakia has committed itself to continuous safety improvements of NPPs, as demonstrated by its ratification of the Convention on Nuclear Safety. In fact, the new Slovak legislation adopted in years 2004 - 2006, in particular the Atomic Act No. 541/2004 Coll. and the Decree No. 50/2006 Coll., requires taking into account the state of the art in the area of nuclear safety, including implementation of design modifications and relevant procedures addressing severe accidents. These new requirements are being addressed in the construction of new plants and safety upgrading of existing plants in Slovakia.

On the basis of this overview, the current paper consists of two parts. In the first part, the paper analyses and presents comparison of current international requirements (IAEA Safety Standards, WENRA Reference Levels, EUR) and Slovak national legislation. From the comparison it is shown that there is good consistency between WENRA Reference Levels, IAEA Safety Requirements, and Slovak legislative documents. It is found that the WENRA Reference Levels are reasonably-balanced requirements, applicable for both existing and new designs, as compliance with WENRA also ensures full compliance with the IAEA Safety

Requirements for design as well as with Slovak legislative documents. Such compliance is also true for relevant IAEA Safety Guides [3, 4] in the parts relevant for existing reactors. Similar recommendations are found in IAEA Safety Guide [2] and EUR. No contradictions were found although there are some differences in terminology and significant differences in the level of details (more details are included in the EUR). It can be stated that, as far as safety-related requirements are concerned, no major deviations of IAEA Safety Standards and Slovak legislation from EUR were identified and that, for all deviations, certain alternative measures can be considered to address the relevant safety issues.

In its second part, the paper evaluates the possibility of safety upgrade of VVER 440/V213 units, in order to comply with WENRA Reference Levels related to severe accidents. Severe accident mitigation features were not included in the original design of VVER 440/V213 NPPs, but the current trend is to eliminate this design weakness. Relevant measures for corium stabilization, hydrogen management, source-term reduction, containment overpressure protection, and long-term heat removal are discussed in the paper by using Mochovce NPP Units 3&4 (currently under construction in Slovakia) as the reference plant.

It is concluded that the implementation of such measures is feasible with present-day engineering practices, allowing to reduce the radiological risk from severe accidents to the level required for current reactor designs. In addition, all the hardware measures implemented in the Mochovce Units 3&4 appear feasible also for existing VVER 440/V213 units. In view of the expected long-term operation of existing Bohunice V-2 and Mochovce 1&2 units, the upgrade of their safety level also in the area of severe accident management is being considered. In the definition of the priorities for the implementation of selected measures several factors need to be taken into account, such as the contribution of these measures to the risk reduction, the complexity of their implementation, the realistic timing for preparation of investments, the existing system and rules for procurement and investments, and the investment cost.

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Safety Assessment and Improvements in Indian NPPs

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Nuclear Power Corporation of India Limited (NPCIL) is the organization responsible for site selection, design, construction, commissioning, operation, maintenance, plant life extension and decommissioning activities of nuclear power plants in India. Presently, NPCIL has 17 Nuclear Power Plapts (NPPs) in operation with a total installed capacity of 4020 MWe and is constructing 5 NPPs, which will add 2660 MWe. With rich experience in all relevant fields of nuclear power generation and to meet the emerging regulatory requirements, safety of Indian NPPs is continuously being reviewed and upgraded. The paper will detail some of the safety improvements and activities for the existing NPPs In India.

Life extension and Safety Up gradation of BWR TAPS-1&2

Tarapur Aiomic Power Station (Units 1&2) is a twin-unit BWR plant with an installed capacity of 2 x 210 MW(e). Commissioned in 1969, this was the first nuclear power station in the Indian sub-continent and is one of the longest serving Boillng Water Reactor (BWR) plant in the world. A comprehensive review of the plant including station operating performance, ageing assessment & management, design basis & safety analysis and structural integrity studies, after 30 years of operation using relevant current safety standards and practices concluded that the physical condition of the station permits continued operation for several more years.

The salient system modifications, which were identified as part of this review process and have been implemented inciudes:

— Upgradation of existing 3x50% capacity Emergency Diesel Generators by 3x100% capacity Emergency Diesel Generators.

— Segregation of electrical distribution system for Class-III (415 V AC), Class-II (120 V AC) & Class-I (250 V DC & 48 V DC) supplies into two zones with physical barrier.

— Redistribution of supplies to redundant loads from separate buses.

— Cable re-routing through diverse routes for redundant loads.

— Provision of supplementary control room.

— Augmentation and unit wise segregation of Emergency Feed to reactors by addition of one pair of Control Rod Drive Hydraulic pumps.

— Unit wise segregation of Reactor Shutdown Cooling System and de-linking from

- Spent Fuel Pool Cooling system.
- Upgradation of Compressed Air System by providing additional dryer and
- powering of compressor by Class-III power supply
- Installation of strong motion seismic instruments.
- Fire Protection system upgradation.

Retrofitting of Components and Upgredatlon for old PHWR units:

Rajasthan Atomic Power station (RAPS) and Madras Atomic Power Station (MAPS) are the earlier generation Pressurized Heavy Water Reactor (PHWR) In India. The design philosophy of later PHWRs has undergone continuous improvements. While it is not possible to implement all the new concepts but certain modifications and retrofitting are incorporated in these old units to bring their safety standards close to that of new plants.

These include;

Retrofitting of high pressure ECCS, installation of supplementary control room, segregation of power/control cables, provision of dedicated instrument air supply to safety related valves, Upgradaition of thermal shield cooling, incorporation of PC based DNM system, Modification in CTM system for replacement of transmitters, recorders, replacement of. main generator excitation with static type and replacement of 230 KV ABCBs with SF6 breakers, replacement of Cl-III/II switchgear and replacement of generator stator winding.

Improvements for facilitating system maintenance

Active Process Water (APW) system to cool plant systems and auxiliaries is provided as a unitized system in Tarapur Atomic Power Station (PHWR Units 3&4). Maintenance on APW system requires shutdown of this system. As shut down heat exchangers are fed by APW system; during APW system shutdown cold shutdown state cannot be maintained. On the basis of safety analysis a design provision is made at TAPP-3M to interconnect APW system of both units in such a way that one unit in addition to its own APW requirement can provide limited supply to shutdown heat exchangers and spent fuel storage bay of the other unit. This configuration permitted taking up APW shutdown after 7 days of reactor shutdown without compromising safety.

Accident Management

Indian PHWRs have event based Emergency Operating Procedures (EOPs) to handle accidents within the design basis, These procedures are being complemented with the symptom based guidelines, which are developed to maintain identified 'safety functions' - monitored continuously through the identified set of plant parameters (symptoms) and maintained in the acceptable state.

Accidents beyond the design basis resulting in core damage are managed through the severe accident management guidelines. These are under development for Indian PHWRs, wherein inventories of heavy water moderator in calandria and light water in calandria vault have an important role in delaying/restricting progression of core damage. The full set of severe accident management guidelines include entry and exit criteria with severe accident guidelines. These guidelines address the core damage states that are likely to result from the progression of severe accident including the threat to containment, the final barrier to the radioactivity release.

One of the important aspects of accident management is human response during the accident progression, which depends on control room ergonomics, procedures, training and stress level of the operating crew. In order to assess and ascertain optimum human responses during the events, systematic studies of human response on plant simulator are being carried out and the outcome of these studies is positive.

Experience Feedback of Current LWR on the Design and Operation of Advanced LWR, under the Safety Analysis Point of View

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Current LWR are operating with an excellent safety record, and it is anticipated that they will be gradually replaced by more advanced reactors (Generation III, III+ and IV). So far, the industry has accumulated a large experience in safety evaluation and operation, and the actual limits of the current reactors and fuel designs, in terms of strengths and weakness, are well known today. Newly proposed designs for Generation III and III+ are the strategic answer to this state point. In all cases, very improved safety features have been incorporated resulting in an impressive reduction of three orders of magnitude in the probability of sever core damage and release of radioactivity respect to current designs. However the approach given by the vendors is different. While EPR relies in the extensive utilization of robust proven equipments, with extended redundancy of structures and active systems, AP1000 moves to the simplification in the number of valves, pumps, loops and auxiliary systems. ESBWR design is conceptually a step forward since gravity is credited for and plays a fundamental role both in steady state and during accidents. In all cases, advanced instrumentation and large and robust vessels are provided. Generation IV reactors are still in the conceptual stage and they still have large room for modifications and improvements. There reactors take benefit of old and new ideas and try to reach greater thermal efficiency, better utilization of the fissile content of the fuel, under the concerns of non-proliferation policies.

Although advanced LWR are focused in providing an adequate response to severe accidents, consideration has to be taken to less unrealistic events, that is abnormal operational occurrences and bounding accidents. In this sense, the main purpose of safety analysis is to guarantee defense to society and the environment against any harm due to nuclear plant operation. Based in this analysis for normal and abnormal operation, the necessary equipments to minimize radiological risks are established and maintained. Therefore, before any modification in the reactor coolant system is performed, it is necessary to review the supporting safety analysis of record to ensure that there is not increased radiological risk in the case of an accident.

Great modifications to the RCS have been carried out so far, such as steam generator replacement in PWR, equipment modification or substitution, or minor modifications, such as partial steam generator tube plugging, surge line redefinition, or other. In other cases, it was necessary to understand unexpected events, not only to verify that the plant was operating within the safety margins, but also to show how to proceed in order to continue the safe operation, after an event. In these cases, a review of the accident analysis was needed.

In the other hand, the nuclear fuel has been substantially evolving, incorporating materials and geometrical changes to improve neutron economy, thermal efficiency and mechanical behavior. All these design modifications required the extensive utilization of thermal hydraulic design methods, in combination or coupled with fuel rod and nuclear design methods.

The accumulated experience in Generation II reactors, actually in operation, leads us to consider the following ideas for the future reactors:

- Safety analysis is the result of the needs of nuclear reactors. We faced changes due to the operating needs (control systems), as well as due to failure of equipments (SG replacement).
- We faced operating deficiencies and they were corrected,
- Safety analysis has served to these main functions. However, could be even more useful in the near term, for instance with a kind of real time assessment?

This review allows us to glimpse some proposals for the future reactors and how the safety analysis should be reshaping for.

Regarding the analytical tools and methods of evaluation, current thermal hydraulic analysis methods were actually developed during the 70´s and 80´s decades, are generally based in the solution of two phase flow version of the Euler and Navier-Stokes equations of conservation of mass, energy and momentum, in transient state, in finite differences schemes with volume averaged properties. The system is actually discretised in multiple connected control volumes with experimentally founded closure relationships for transport of heat, mass and pressure drop. System actuations and environmental influences are specified as boundary conditions. Besides, simulated scenarios and initial conditions are selected to ensuring the conservatism of the result. Depending on the degree of realism of the different basis of the simulation, several approaches, such as superbounding, BEPU and fully best estimate methods, have been utilized. Recently, thanks to the availability of great computing facilities, this macroscopic viewpoint is somehow complemented with the use of advanced analytical tools such as CFD. However, the gap between the macroscopic and microscopic approaches is somehow not entirely established as standardized methods of analysis. FEM, and FVM could play an important role in this intermediate range of detail.

It can be concluded that, Safety Analysis as a knowledge framework will keep being important in supporting operation, not only of life extended current reactors, but also of advanced LWR. In order to face the more demanding analytical needs of next generation reactors, more detailed codes, models and methods will be necessary, although still based on, or not very different from the current state nuclear engineering state of the art. It is foresee that current fuel designs will remain valid for the advanced LWR that will be deployed from next decade, with minor modifications in geometry and materials, well within the current accumulated experience. However, enough room will be left for extensive research of new concepts and developments of more robust features to be incorporated in advanced fuel designs. At all stages, more involvement between utilities, vendors, research facilities, universities, and technological institutes, will be needed in order to guarantee a successful generational replacement and maintenance of the high standards of safety and efficiency characteristics of the nuclear industry.

Operating Nuclear Reactors in Ukraine: Enhancement of Safety and Performance

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Currently fifteen water-cooled water-moderated reactors operate at four Ukrainian Nuclear Power Plants (NPP). producing over 50% of total electricity in the country. Role of the nuclear power in the fuel and energy branch of Ukraine continually increases. The National Strategy of the nuclear industry development envisages extensive construction of new nuclear power facilities during the next 20 years.

In line with the strategical plans for further growth and development of the nuclear power sector, safety enhancement of the nuclear power units in operation in one of the major tasks of the state policy in the sphere of nuclear energy utilization. Being of high priority for the Ukrainian Authorities during the entire period of the country independence, this task has received the utmost and permanent attention within the recent 4-5 years. During this latest period the Ukrainian nuclear industry has implemented wide range of safety enhancement measures at all operating nuclear plants. The effors have been predicated on results of the comprehensive analysis of existing safety issues, identified and categorized in accordance with the IAEA recommendations, with due regard to the operating experience of Ukrainian and similar foreign plants, and in accordance with the established priorities.

Major current industrial activities on enhancement of safety and performance of nuclear plants are being ruled by the “Concept of Safety Enhancement of NPPs in Operation” approved by the Ukarainian Government in December 2005. The Concept, with its scheduled completion in 2010, includes broad specturm of safety upgrade measures that are intended to bring NPP safety to internationally accepted level. It covers several most vital safety enhancement areas and associated safety measures developed using information on risk obtained from the comprehensive safety analyses conducted for the Ukrainian NPPs within the laste ten years (Safety Analysis Reports and international co-operation programs on NPP safety assessment). The distinct safety enhancement measure, which have been included in the Concept, take credit of the qualtative and quantitative outcomes from the recent probabilistic safety assessment studies. The main goal is to further enhance safety of the Ukrainian NPPs to account for enforcements in regulations and best current practices in rational and the most efficient way.

The paper will discuss the priorites, progress and current results of the operating NPPs safety enhancement activites both from the regulatory and industrial perspectives. It will outline the challenges and plans for further operation of the existing Ukrainian nuclear reactors, and discuss the expectations of the national regulatory authorites in view of the planned construction of the new nuclear power plants.

Synopses for Topic 7

INSTRUMENTATION & CONTROL

Upgrading the Reactor Power Control Concept with a Modern Digital Control System

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Within the framework of a retrofit project, a reactor power instrumentation and control system (REALL) — consisting of a limiting system and the respective reactor control systems — was retrofitted and modernized in a 1450-MW-nuclear-power-plant in Baden-Württemberg. The REALL process control functions were implemented within a modern and completely digitized control system that has been designed for use in safety I&C applications. Along with the installation of the digital control system, the associated hardware was adapted to today's state of the art. At the same time, the given potential for improvement, as revealed during the plant's operation so far, was taken into account in the programming.

In order to provide for transparent and quality-assured project management, the implementation was based on a stage plan consisting of several steps, along with specific milestones.

Final commissioning of the modern digital control system took place during the 2008 plant overhaul. Despite the complex commissioning procedure, it was possible to avoid a major prolongation of the plant's downtime and to keep within a rough 4-week timeframe that had originally been defined for the plant overhaul — thanks to adequate structuring of the project, goal-oriented implementation of preparatory infrastructural measures and adequate scheduling of the coordinated activities of the installation and commissioning teams entrusted with the commissioning of the digital control system during the overhaul activities. In addition, consistent application of quality-assuring measures during the entire course of the project considerably contributed to a smooth commissioning process.

Within the digital control system, the REALL functions were engineered and implemented by using a system-specific tool system which included a graphic editor. A series of analyses were carried out with the help of tools designed for redundancy comparison and for simulation and validation of the implemented process control functions after code generation. These served as a means for discrete checking and documentation, and helped to detect project engineering errors at an early stage.

Based on the results delivered by the validation test and on the requirements resulting from process concepts, the plant simulator in Essen was adapted to suit the specifics of the system and the respective peripherals in order to be able to integrate and test the REALL functions implemented in the digital control system at the simulator. In addition, process-related modifications intended to optimize the plant that were made in the course of the project were subjected to a complex transient analysis.

All switchgear cabinets were manufactured in the manufacturer's factory. The factory acceptance test was performed on the test floor on the basis of the works test instructions approved by the official expert. After successful completion of the acceptance test, the pretested switchgear cabinets were installed in the plant in October 2007 during regular load operation, without causing any interferences, and were connected to the power supply system upgraded earlier in the course of a preceding project. In addition, a distribution cabinet was installed in each section of the switchgear building, containing a system connection side and a plant connection side, with intermediary transverse connections. After the installation of the digital control system cabinet had been completed, the system supply cables, which were adapted to match the cabinet's I/O boards, were connected on the system connection side of the distribution cabinet. This way, it was possible to pretest signals, to avoid wiring errors and to considerably reduce the time needed for the rest of the overhaul-dependent cable installation and the associated commissioning tests. In order to furnish proof of the system stability, the digital control system was operated in island mode for several months before the overhaul activities were started.

The new control system was integrated into the overall plant concept during the 2008 overhaul after the old REALL system was shut down. After all the control system commissioning tests were completed, various process-specific commissioning tests followed in order to assure an integral verification of the equipment's proper functioning. These tests included the startup operation of the plant. After the process-specific commissioning tests in August 2008, the plant resumed its power operation. So far, the operating results with the new REALL implemented in this modern digital control system have been very positive.

With the consistent documentation of both the course of the project and the stage-specific tests, which were accompanied by an official expert, it was possible to meet the high quality assurance demands of such a complex system. Despite the large scope of the replacement activities, this project is a very good example of how a complex system retrofit can be accomplished successfully through efficient project structuring, in combination with selective preparatory measures and the respective quality assurance measures, without causing a significant prolongation of a scheduled plant downtime.

Qualification of FPGA-Based Safety-Related PRM System

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Toshiba has developed Non-rewritable (NRW) Field Programmable Gate Array (FPGA)-based safety-related Instrumentation and Control (I&C) system. Considering application to safety-related systems, nonvolatile and non-rewritable FPGA which is impossible to be changed after once manufactured has been adopted in Toshiba FPGA-based system. FPGA is a device which consists only of basic logic circuits, and FPGA performs defined processing which is configured by connecting the basic logic circuit inside the FPGA. FPGA-based system solves issues existing both in the conventional systems operated by analog circuits (analog-based system) and the systems operated by central processing unit (CPU-based system). The advantages of applying FPGA are to keep the long-life supply of products, improving testability (verification), and to reduce the drift which may occur in analog-based system. The system which Toshiba developed this time is Power Range Neutron Monitor (PRM). Toshiba is planning to expand application of FPGA-based technology by adopting this development process to the other safety-related systems such as RPS from now on.

Selected application of FPGA-based I&C systems is PRM of Boiling Water Reactor (BWR). The PRM system monitors reactor power by measuring neutron flux level and issues trip signals when the conditions monitored reached to the setpoints. PRM can be used in safety-related (class-1E) systems. The developed system is compatible with the conventional analog-based systems to minimize the requested cost for upgrading.

FPGA circuit is designed by using Very High Speed Integrated Circuit Hardware Definition Language (VHDL). The process of implementing VHDL to an FPGA circuit is almost similar to the process of implementing application software to read only memory (ROM) in a CPU-based system. For this reason, Toshiba has developed a design and manufacturing method of FPGA application which can apply the technique of Verification and Validation (V&V: specified in IEEE 7-4.3.2) as a qualification procedure for the process of manufacturing FPGA circuits.

After the establishment of the FPGA products design process, Toshiba designs, and manufactures the NRW-FPGA based PRM units according to the design process to confirm their performance, including the Qualification tests including the environmental, seismic and Electro Magnetic Compatibility (EMC) tests.

Design Feature and Prototype Testing Methodology of DHIC's Nuclear I&C System

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DHIC has developed an I&C(instrumentation and control) system for nuclear power plant through Korean Government R&D project since 2001. This I&C system is designed and implemented to apply for new 1400MW Nuclear Power Plants of KHNP in 2009.

This system design is based on class-1E PLC platform and non-class1E DCS platform. Class 1E PLC's features are high speed processing capability, CPU redundancy, safety data link for communication between safety channels, safety data network for intra channel information communication and redundant power module. Also, the communication speed of data link and network is up to 16M bps. PLC qualification tests such as environmental, electromagnetic interference and seismic testing were conducted. Safety evaluation report for PLC topical report issued from Korean Reulatory Body. Non-Class-1E DCS provides redundant CPU/rack configuration, separated control and information networks and other necessary functions such as function blocks. DCS's network for control and information is redundant and its communication speed is 100M bps. Communication link between redundant CPU's communication speed is up to 10M bps.

RPS(Reactor Protection System), ESF-CCS(Engineered Safety Feature-Component Control System), CPCS(RCOPS, Reactor Core Protection System) and safety grade monitoring system are designed, implemented and tested. All of software life cycle activity was documented and verified in conformance with IEEE Std.1074 and IEEE Std.1012. Safety evaluation report for RPS, ESF-CCS and RCOPS topical report issued from Korean nuclear reulatory institute. These systems meets Defense in depth and diversity and have simpler structure, superior ability in maintenance and self-diagnostics and more advanced safety algorihthm than previous systems used in Korea. Various system tests were conducted on these systems. These include response time test , abnormal input conditions test and equipment failure modes test which cover all redundancies, loss of power, CPU failures, data communication failures, power supply failures, card removal, etc. For example, the response time requirement of RPS is within 150 msec.

PCS(Power Control System), P-CCS(Process-Component Control System), NPCS(NSSS Process Control System), NIMS(NSSS Integrity Monitoring System) and non-safety grade monitoring system are designed, implemented and tested. DHIC has built the prototype test facilities to meet proven technology requirement according to the requirement of KURD(Korea Utility Requirement documentation). These systems are composed of MCR facilities, APR 1400(Advanced Power Reactor 1400) code simulator and the developed I&C systems. MCR facilities consist of Large Display Panel to display the operational status of the NPP, operator consoles and a safety console (Fig1). These are fully digitalized facilities, which have the touch-screen control method.

The integration level test such as interface test, functional test, response time test, network load test and operation test was performed using the prototype test facilities.Various response times between developed systems were calculated and measured. The response time from input of tested system to MMI display was calculated and measured for protection signal, control signal, information signal. The test result meets the design requirement. For example, the response time requirement of information signal is within 2500 msec. Network load test was performed to identify network performance would endure simulated full load condition as simular to plant real operation status and the performance test result was satisfied. Also operation tests were performed in order to acquire operational historic data. Operation test include load rejection, load transient, load cycle, reactor power cutback, FWCS (Feedwater Control System) valve transfer test of reactor power increase process.

Through thorough development procedure, evaluation, testing and V&V which meet code and standard(ex: IEEE and KEPIC) and KURD, DHIC's new I&C system is going to be applied for new 1400MW NPP in 2009.

Fig.1 Main control room overview of performance test facility

FPGA-based Technology (Systems) for I&C of Existing and Advanced Reactors

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Control systems of modern nuclear installations (including water-cooled, WCR) are based on programmable technologies. Most of control systems modernizations which are implemented at operating nuclear installations are also based on application of programmable technologies.

Besides, a range of features and properties is defined for programmable technologies. These features and properties make licensing process more complicated, facilitate appearance of common cause failures, make safety evaluation procedures more complicated, etc.Also it is known that programmable technologies significantly extend the time periods for project realization of new power units construction and modernization of the existing power units, and also it involves rise of its value.

Company RADIY has developed the Platform of digital equipment RADIY on FPGA-based technology.In the article there is a description of the features of FPGA-technology developed and applied by Company RADIY, features of the Platform RADIY and systems realized on its base, which allow to minimize significantly above-mentioned negative features and properties of programmable technologies.

Technology which realized in Platform RADIY allows to solve the whole set of tasks of control (including regulation) and protection of nuclear installations.Platform RADIY is a combination of the best features of traditional programmable technologies and FPGA-technology.

Peculiarity of Platform RADIY is the fact that all safety-critical functions of control system are realized as a “hard logic” on FPGA chip. Software is used only for auxiliary tasks.

Platform RADIY is approved since 2004 through operation of more than 40 up-to-date digital control and protection systems at 15 power units of nuclear power plants of Ukraine and Bulgaria.

According to the opinion of the authors of this article the technology which is realized in Platform RADIY is the key factor for solving of control and protection tasks of nuclear installations in the nearest future.

On-Line Monitoring and Calibration Techniques in Nuclear Power Plants

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As the current generation of nuclear power plants have passed their mid-life, increased monitoring of their health is critical to their safe operation. This is especially true now that license renewal of nuclear power plants has accelerated, allowing some plants to operate up to 60 years or more. Furthermore, many utilities are maximizing their power output through uprating projects and retrofits. This puts additional demand and more stress on the plant equipment such as the instrumentation and control (I&C) systems and the reactor internal components making them more vulnerable to the effects of aging, degradation, and failure. In the meantime, the nuclear power industry is working to reduce generation costs by adopting condition-based maintenance strategies and automation of testing activities.

These developments have stimulated great interest in on-line monitoring (OLM) technologies and new diagnostic and prognostic methods to anticipate, identify, and resolve equipment and process problems and ensure plant safety, efficiency, and immunity to accidents. The foundation for much of the required technologies has already been established through 40 years of research and development (R&D) efforts performed by numerous organizations, scientists, and engineers around the world including the author. This presentation provides examples of these technologies and demonstrates how the gap between some of the more important R&D efforts and end users has been filled.

Part 3

Synopses for Poster Presentations

Nuclear Power and Ghana's Future Electricity Generation

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One of the major challenges facing Ghana in her developmental efforts is how to meet the increasing electricity demand. Ghana's electricity generation system depends heavily on hydro power which accounts for 68% of total installed capacity. The remaining is taken by thermal power systems. The heavy dependency on hydro systems has led to shortfall in power supply in case of drought. To deal with this situation the necessary steps are being taken to build more thermal plants to complement the hydro systems. The thermal plants currently run on imported light crude oil but steps are being taken to run them on less expensive gas imported from Nigeria through the West African gas pipeline. The conflicts in the Niger Delta, the source of the gas has threatened the security of gas supply and this coupled with the fact that gas price is indexed to that of crude oil have raised concerns about the supply of gas from Nigeria.

This paper presents the results of the assessment made in the Ghana electricity generation system and the role of nuclear power in Ghana's energy mix using MAED projections and the MESSAGE model. This assessment forms part of the IAEA-TC project "Planning for Sustainable Energy Development in Ghana" which is meant among other things to develop a sustainable energy mix for the country.

Energy projections made by using the MAED model have shown that Ghana's electricity demand expected to increase to about 4000MWyr in 2030. This expected electricity demand far exceeds the total electricity supply capability of the existing hydropower system, untapped hydro resources and the maximum amount of gas that can be imported from Nigeria through the West Africa pipeline.

Technological assessment on the suitability of the various nuclear power technologies has been done based on the grid size, technological maturity, passivity and standardization of reactor designs and it has been found that a water cooled SMR with capacity not exceeding 400MW(e) is the most favorable choice. The economic competitiveness of nuclear power in comparison with other candidate plants under consideration which are coal and liquefied natural gas are highlighted.

The paper also discusses the major issues of concern in connection with utilization which are plant safety and waste management. In addition the main challenge facing the introduction of nuclear power, which is funding is also discussed.

The Role of Nuclear Energy in the European Energy Policy

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One of the strongest messages to emerge from his presentation is the need to extend the lifetime of existing nuclear power plants and build the next generation of reactors that will continue to underscore nuclear energy's important contribution to the goal of achieving a low-carbon economy. Thus *nuclear energy* and the developing Europe's *low-carbon economy* reflect one of the EU's main policy priorities, namely to encourage sustainable economic growth while at the same time reducing the Community's carbon footprint. Without nuclear energy the EU would never reduce its carbon footprint as renewable energies alone could never achieve this goal.

The aim is to give emphasis to the process how the debate in favour of nuclear energy as a main pillar of the fight against climate change has gained considerably in impetus in the European Parliament and to note the great recent strides that have taken place within the Parliament with regards to the nuclear debate.

The key-points of the presentation:

— **increasing problems of supply**

Giving a bleak overview of spiralling global energy demands and stressing the urgent need for new technology, nuclear power and changes in lifestyle to lead the drive to reduce CO_2 emissions by underlining how nuclear energy is an attractive option when it comes to reducing CO_2 emissions and how the cost of carbon abatement is low in comparison with carbon capture and storage and most renewables.

— **global warming**

Underlining what all industrial sectors must do in order to reduce global temperatures by 2°C by 2030, and also speaking about the cost and opportunities of various CO_2 abatement measures, including carbon capture and sequestration.

o **the major pillars of the European energy policy-making: Kyoto-Coppenhagen; Lisbon; Moscow-Riyadh**

Giving an overview of the EU energy policy in general and the importance of the nuclear component in that policy.

Nuclear civil energy is currently being developed everywhere in Europe and in the world, projects of new nuclear power plants are launched, however their construction and operation in nuclear "emerging" countries cannot occur in any conditions. With the nuclear revival well under way in several European countries and ambitious new build programmes in the pipeline, another main subject is the question of supporting the current drive for new build that is gathering momentum across Europe and beyond.

Presentation of the latest policy developments in the nuclear field, including the European Nuclear Energy Forum and the Sustainable Nuclear Energy Technology Platform by stressing the important role that nuclear energy has to play in the EU's future energy mix and the need to guarantee safety and security safeguards when it comes to operating plants and handling radioactive waste.

- **the role of nuclear energy in the 20-20-20 perspective**

Emphasising how the building of a new power plant should be viewed as part of a long-term commitment: A nuclear project covers a period of almost a century – it takes10 years to build a plant, the plants operates for 60 years and the decommissioning phase takes 20 years, also underlining how the technology need to safely store radioactive waste of all levels exists in Europe and that the problems and dangers generally associated with waste can be easily overcome; the public can be convinced of the safety and efficiency of waste storage if the success of operations is communicated effectively.

- **European knowledge and R&D in the nuclear field**

Stressing the urgent need for governments and academia to do more to identify, recruit and retain the most talented young scientists and researchers -because it is they who will sustain the nuclear revival. Summarizing the debate focusing on a number of initiatives launched by the nuclear industry to redress the "nuclear talent deficit" and of the gradual improvement of the situation in a number of countries.

The industry should do to attract more young people to pursue a career in nuclear engineering and research, also should stress the importance of skills building and innovation as ways of attracting young people and of the need for knowledge transfer from the "older" generation to the younger generation.

Summing up the needs of the research sector by saying that R & D is like building a cathedral, it requires faith.

- **the perception of nuclear energy in the European Parliament**

In technical matters relating to nuclear energy, the EURATOM Treaty gives the European Parliament a merely consultative role, thereby limiting its political influence on the Council. However, with the strengthening of the codecision procedure, this has no longer been the case for economic matters since 2004. Currently the EU is in the process of defining its economic strategy for the decades to come, and fundamental decisions for our future are to be taken on the legal basis of the EC Treaty, under which the Parliament is co-legislator. After presenting these processes and unveiling that one of the strategic decisions to come in the near future in the European Parliament is the definition of Europe's energy policy, the author will endeavour to show that nuclear energy shall necessarily be a substantial part of that policy.

We cannot exclude the use of any energy resources just for political or ideological reasons. The key factor which must be regarded when composing the energy mix is environmental protection and the mitigation of the effects of climate change… Nuclear energy is a major contributor to the security of electricity supply.

Speaking out in support of nuclear, highlighting broader issues of employment and citizens' access to inexpensive electricity as her main concerns.

- *Social perception and public acceptance. economic and social impacts of nuclear energy*

Nuclear energy is attracting renewed interest due to its potential but plays a disputed role in mitigating global warming and, more generally, in supporting sustainable development. Although its public acceptance could be largely improved if the positive role of nuclear energy was clearly demonstrated and better communicated.

- *Civil society expectations regarding the transparency of nuclear energy*

Nuclear energy is among those industrial activities that face high expectations for transparency and accountability in decision making. Despite important cultural differences across countries, a number of common features characterise media and public expectations regarding risky activities. The major aim is to identify a common understanding of the main stakeholders' expectations on the conditions and practices which could improve the transparency of nuclear regulatory activities.

- *Civil society involvement in the nuclear decision-making process*

The public needs access to all relevant information that is available and that is not considered too sensitive for reasons of national security or commercial confidentiality. A high degree of trust and transparency needs to be established and maintained with the public who have the right to be part of the decision-making process in the nuclear energy domain.

Challenges and Opportunities to Launch of New Nuclear Power Programmes in India

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The biggest challenge for production of Nuclear power has itself become the biggest opportunity for its advancement.

Nuclear power's biggest challenger had been the cost aspect of cheaper fossil fuels. This itself is biggest opportunity because of climate change & global warming that is being driven primarily by emission of man-made green house gases due to fossil fuel burning.

With the world understanding the pressing need to conrol green house gas emission & resultant global warming, the search for alternative means of energy has opened flood gates for various competitive technologies, nuclear among them being fore-runner in near future.

Undoubtedly drive for development at Nuclear Power had come from an unquestionable belief for technological innovation in past but now we are standing at crucial juncture of world history wherein our next few steps would decide the future course of humanity.

Energy perspective cannot be seen only in the national interest and it has to be seen in the international perspective with regards to its environmental impact without obstructing the right of developing countries to meet their basic needs.

In view of above facts

Opportunities:

1. Despite the world (with about six and half billion people) advancing at the break neck speed, only about a billion and half enjoy the modem energy. Thus there is lot of potential demand.
2. With the world population increasing every minute, demand for energy can do nothing but grow. So even maintaining the present nuclear power share of 16% in itself is huge demand for nuclear power.
3. Climatic changes related to global warming due to fossil fuel burning has made the world think whether continuing with present trend would be beneficial in long run and what we would leave for our future generations.
4. Impact of Climate Change policy and growing acceptance of Carbon credit & Carbon Cap concepts. Understanding that even Bio-fuel is not a green fuel as thought initially.
5. Phenomenal demand visibility is there if share of nuclear power is to be increased beyond 16%.
6. With depleting reserves and increasing costs of fossil fuels would shift the economics in favour of Nuclear Energy.

7. Recent oil price spike has forced the nations to seek for diversification of energy resources for the sake of energy security.
8. The countries going nuclear first time would get access to latest technology. This would help up gradation of other industries also.
9. With respect to fossil fuels, the relative nuclear fuel availability can be essumed unending.
10. Advancement of technology of Space research & defence research can be fine tuned for nuclear power plants.
11. Technological exchange between the countries & operating experience sharing through IAEA, WANO, INPO, COG etc has resulted in significant improvements in nuclear plant reliability & a progressively improved safety record in Past two decades (since Chernobyl disaster in 1986).
12. Willingness of nuclear power capable countries to share technology.
13. Quantitatively the waste generation of nuclear power is around 100,000 times less than fossil fuel plant so correspondingly the energy & space requirement for disposal is also much less but...

Threats:

1. Nuclear waste despite quantitatively being much less than fossil fuel plant is highly radioactive. Handling & safe disposal of the same.
2. Radiation has many harmful effects, many of them affect genetically, so the built in safety systems / mechanisms are to be strengthened with defence in depth approach.
3. Connection between nuclear power & nuclear weapons exists because both require Fissile material. So fool proof audit system is to be evolved to ensure that none of fissile material is diverted out.
4. Significant improvements are needed to ensure Physical security of Nuclear material world wide because great risk pops up if terrorists acquire nuclear, or radioactive material.
5. Long gestation periods and high start up costs in present era of Economic recession & non availability of credit.
6. Regulations related to nuclear industry need more uniformity, teeth & acceptance by all.
7. Innovation, both in terms of technology & policy.
8. A fast emerging prime concern is the security of Nuclear Power Plants against the terrorist attacks.
9. Building adequately trained Human infrastructure for projected growth of nuclear energy is monumental task.
10. To foster adequate support by the national industry so that targeted level of participation in execution of project is attained.

Indian Perspective:

a. Presently Nuclear power is the exclusive responsibly of Government, Formation of policy frame work to have private sector participation is in progress.
b. Presently 17 operational plants, 6 under construction & many being planned, Recruitment, training & retaining the adequately trained manpower.
c. Ensuring the continuity of fuel supply for above.
d. Achieved success with fast breeder test reactor technology & proceeding with Prototype Fast breeder reactor.
e. The increased production of radioactive waste will have to be managed, taking into account safety requirements & the fact that India has relatively higher population density.

Nuclear Power Project in Thailand

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Thailand has been highly relied on fossil fuels for electricity generation. In fact 66% of today's electric power is supplied from natural gas. With current unprecedented increase of oil and gas prices, the country is in need of alternative energy sources more than ever. The Government recognizes the problem and seeks sustainable solution not only to improve energy security but also to reduce greenhouse gases emission, the root of threatening global warming problem. For base load power generation, however, nuclear power is perhaps the only practical option currently available. As a result, in Thailand Power Development Plan 2007-2021 (PDP 2007), there will be a 1,000 MWe nuclear power plant commercialized in 2020 and another in 2021. By the end of 2021, nuclear share of electricity generation of Thailand will be about 5%.

Due to the fact that this is Thailand's first nuclear power plant, necessary infrastructures are not currently in place. To cope with this requirement, on April 11, 2007, the National Energy Policy Council (NEPC) appointed the Nuclear Power Infrastructure Preparation Committee (NPIPC) to develop the Nuclear Power Infrastructure Establishment Plan (NPIEP). NPIEP comprises two major plans: nuclear power infrastructure and nuclear power utility preparation plans. Required infrastructures include: legal and regulatory systems and international commitments; industrial infrastructure and commerce; technology development and transfer and human resources development; nuclear safety and environmental protection; and public relations and public acceptance. Utility planning comprises preparations for setting up organizational structure to accommodate a nuclear power project, technology selection, assessment of nuclear safety and technical aspects of nuclear power generation, and implementation of project feasibility study and site selection. NPIEP had been effectively developed under guidelines and technical support from the Intemationai Atomic Energy Agency (IAEA), On December 18, 2007, NPIEP was approved by the Cabinet, including implementation plan and budget for 2008-2010. Furthermore, Nuclear Power Program Development Office (NPPDO) and Nuclear Power Infrastructure Establishment Cooperation Committee (NPIECC) will be established as parts of NPIEP.

During 2008-2010, pre-project activities will be carried out. These activities include infrastructure establishment, utility preparation, and public education and participation. Electricity Generating Authority of Thailand (EGAT), a government owned utility under the Ministry of Energy, will be mainly responsible for utility preparation works, i.e. survey and evaluation of potential sites, project feasibility study, initial environmental examination (IEE) and human resource development for utility. NPIECC will submit the readiness report, including status of infrastructure preparation, utility preparation, and public education and participation, to the Cabinet for approval by the end of 2010. According to the nuclear power project schedule, nuclear power project will be implemented during 2011-2013, construction will start in 2014, and Thailand's first nuclear power plant will be complete for commercial operation in 2020.

Today's main concerns in energy policy and planning are how to decrease dependency on high priced oil and gas, and how to reduce greenhouse gases emission. Like many countries, Thailand is considering a nuclear power option especially for a large scale base load plant. Nuclear power plant does not emit greenhouse gases due to the fact that no combustion taking place in power generating process. Nuclear technology is a proven technology and the industry has continuously improved efficiency and safety of the plant. Nonetheless the public education on major issues such as nuclear safety and radioactive waste management must be carried out continuously.

Economy Aspect for Nuclear Desalination Selection in Muria Peninsula using 1000 MWe PWR

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An assessment of economy aspect for nuclear desalination selection has been carried out. This study is done to explore any possibility to utilize co-generation concept of desalination, because there is a plan to introduce nuclear power plants (NPP) into Indonesia's electricity grid.

A comprehensive study on different energy sources shows that NPP is economically and technically viable to be introduced into the grid in 2016/2017. The candidate site is Muria Peninsula in Central Java. Currently, the total of install electricity capacity is about 29.083 GWe, and it is estimated that electricity energy growth is about 7.1% per year. The install capacity in Java is about 23 GWe (65% of national capacity). With economic growth projection is about 6%, therefore in the 2025, it is needed electricity energy about 70 GWe, so electricity demand increase 2000 MWe per year. Therefore, a PWR of 1000 MWe will coupled with a desalination plant of MSF (Multi-stage Flash Distillation), MED (Multi-Effect Distillation) and RO (Reverse Osmosis). The costs of water production for the Multi Stage Flash Distillation (MSF), Multi Effect Distillation (MED) and Reverse Osmosis (RO) desalination process coupled to PWR 1000 MWe would be compared.

The objectives of the economic evaluation is to help the decision-maker to eventually implement an integrated nuclear desalination plant, generating both electricity and fresh water. Economic analysis of water cost are performed using a computer program issued by the IAEA, DEEP-3.1. In this study, option for turbine scheme is set as extraction and backpressure. Options for specific carbon tax, thermal steam compression and backup heat are not used. Construction cost for NPP is assumed to be 2600 \$/kW, production capacity 2.750 m^3/d, interest rate 5%, construction cost for MSF 1200 \$/m^3/d, MED 900 \$/m^3/d and RO 700 \$/m^3/d, ratio of recovery RO 45%, top brine temperature for MED 65°C and MSF 110°C.

The results of the performed case study of this Muria Peninsula showed that the water cost to desalination process coupled with PWR nuclear power plant (at 5% interest rate, 2750 m^3/day capacity, 28°C temperature, 28.700 ppm TDS) with MSF plant is the highest (1.353 \$/m^3), compared to 0.885 \$/m^3 and 0.791 \$/m^3 with the MED and RO plants. As for MSF process, water cost by RO are also sensitive to variables, such as the interest rate, temperature and total salinity. However, MED process are sensitive only to interest rate and temperature. An increase in the interest rate of a certain value will increase the water cost produced by PWR+MSF installation more than others. An increase of seawater temperature increases water cost of MED and MSF, but it decreases the cost of RO. An increase of total salinity causes water cost increase in MSF and RO. Water cost of MED is not affected by total salinity at all. Back-pressure turbine scheme produce cheaper water than that of extraction. It can be concluded that MED desalination process can be a good alternative for PWR nuclear power plant in Muria Peninsula based on economy aspects.

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Infrastructure and Other Considerations to Launch Nuclear Power Programme: the Case of Sub-Sahara African Developing Countries Like Ethiopia

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Trends in the world's population and energy use during the past century show dramatic and relatively parallel increases in both [1]. The demand for electricity is expected to increase more rapidly than the demand for other forms of energy and nearly double by 2020 [1]. I. Kimura [2] also remarked that: *"For welfare and prosperity of mankind and even for world peace in and after the next century, vast amount of energy is required."* This worldwide growing energy demand and the rising concern of climate change led to the need for production of significant amount of 'safe and clean' energy which in turn favours to nuclear option. Other alternative renewable sources like solar and wind can assist but currently they are short of supplying the required high energy demand either economically or/and in substantial amount [3]; other than their high dependency on weather. Nuclear option therefore remains a possible (developed) technology to fill this energy gap in better conditions of safety and sustainability without large greenhouse gas emissions. [4].

Vast amount of potential energy resources including hydro, coal and uranium reserves exist in the Sub-Sahara African region; yet it is very deprived of energy provision and is one of the least developed part of the world. One reason to the region's under development can therefore be attributed to lack of adequate energy. As Dr. ElBaradei also once remarked: lack of energy in developing countries is one of the highest impediments for development and the fight against poverty [5].

The socio-economic situations of most Sub-Saharan countries are similar; and mainly based on subsistence farming. The electric power generated in the region is mostly from hydropower stations; with few oil generators and coal plants. Currently, in most of these countries there is very high shortage of electricity and continuous power outage due to repeated draught and shortage of water reserves in dams resulting in shift supply system; which in return lead to drop of GDPs and reduction in income of small businesses and families that worsen the existing poverty.

For instance, Ethiopia is one of the Sub-Saharan country in the Horn of Africa; with a population of about 77 million. The country's most important, developed and high potential source of energy is hydroelectric power. The Ethiopian Electric and Power Corporation (EEPCO), a government corporation, operated most of the country's power systems. EEPCO incorporated all electric power stations and currently generate about 875 MW per day, which is short of 150 MW at peak times [6]. Ongoing efforts to increase its production by 3270 MW are expected from operational hydropower projects. The energy demand of the country in recent years also increased annually up to 17% [7]. There is also interconnection of electricity grid projects, East African Power Pool (EAPP) among four East-African countries in the sub-region; Ethiopia, Djibouti, Sudan and Kenya [7].

According to news from IAEA and the World Nuclear Association nearly twenty countries are actively considering embarking upon nuclear power programs [8]. These ranges from sophisticated economies to developing 'new' nations. Almost all North African countries; Nigeria and Ghana from West to Namibia in the South are included. Some reasons for this shift to nuclear options are: to overcome with natural disasters (for instance effect of cyclic draughts); decline of national fossil fuel reserves; and strategic utilization of national resources. Though numerous the reasons may be, it seems that there is growing interest of nations (mainly developing countries) to nuclear power, which may raise concerns of the proliferation of nuclear weapons and safety issues. It is not only electricity and heat nuclear technology provides to the world, but many diverse social and economic rewards; and various spin-off benefits. Therefore, it is sensible and reasonable that nations show great interest towards this technology. Yet, it should be related with all responsibilities that the technology entails and with full commitment of states for its peaceful applications as well as developing all other prudence capabilities that the technology requires.

The Sub-Saharan countries are new to launch a nuclear power (NP) programme. If they are interested to consider this technology, they should be highly committed to develop the required basic infrastructure in stages [9]; and should conduct important activities that need to be completed in phases [10]. This include longer than 100 years of maintaining a sustainable national infrastructure throughout its operation, decommissioning and waste disposal [10].

The major challenges to launch a NP programme in these countries are; lack of funding, inadequate technical know-how, lack of information on the available resources, low grid capacity of nations, lack of required organizations and physical component of the infrastructure. However, there are also encouraging aspects such as commitment to expand electric supply to rural areas, strategic shift to diversify energy sources, availability of uranium (thorium) reserves, availability of basic regulatory infrastructure in radiation protection and nuclear safety, and enhanced regional and international economic cooperation.

In conclusion, the high level of poverty in Sub-Saharan countries mainly is due to lack of adequate energy and its poor coverage. It is vital to assert here that provision of sustainable and sufficient amount of energy in the region can greatly advance development, alleviate poverty and ensure stability. Besides, to come out of this cyclic challenge; countries based on regional economic cooperation and ideals of African Union, should interconnect their electricity grid like EAPP and commonly invest to launch NP programmes in relatively stable countries. Candid support of the international community is crucial, and IAEA should support and encourage such arrangements.

Developmental partners also should play their role of fighting poverty, by availing the required funds, technical assistances, and encouraging their companies in providing insurance and lending funds to this end. This can be a commitment of the developed world to fight global climate change and which is the objectives of sustainable development issues. Added to this, consideration of the region's contribution of the long term supply of uranium to their developments should also be taken into account. The idea of the International Fuel Bank can also be an important initiative that can complement such proposals. However, these countries should use the technology in responsible manner, maintained by effective implementation of international safeguards regime to ensure the concerns of the international community.

Moreover, if the Sub-Saharan countries go for nuclear then the best candidate would be the worldwide dominant water cooled reactor. This reactor types developed through evolution with numerous global experiences in operation and design; which are inherently safe, self-regulated and efficient designs, like AP1000 and EPR. However, for countries with low grid capacity and to assist remote communities or to carry out rural projects which are far-away from national grid systems, considerations for smaller reactor designs with affordable capital cost is required. In this regard designers should be encouraged to invest in R& D.

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The Future of Nuclear Power in Portugal

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1. **Introduction.** The paper will be an update of the paper entitled “Nuclear technology and energy in Portugal” prepared for the IAEA Workshop**:** "Steps for Conducting Nuclear Power Plant Technology Assessments" which took place in Vienna on 17 - 20 November 2008.

2. **Update on the ITN.** The ITN continues to be the main institution in the country dealing with nuclear applications. A probable visit of the author to this institution before the conference will enable to supply more information, particularly on its readiness to develop capabilities to enacting the legal regulatory framework for nuclear power generation and the capacity to implement it.

3. **Energy Policy, Supply and Demand in Portugal**. An Update on the Energy Policy, Supply and Demand in Portugal will be supplied with the most recent figures extracted from the International Energy Agency in Paris and national sources**.** The present Administration continues to place emphasis on renewable forms of energy, without considering the nuclear option in the energy mix. The prospects for nuclear energy in Portugal have been affected by the ongoing economic crisis. This crisis has particularly affected the economy with has a low per capita GDP by European standards and is still too dependent on export oriented industries with a weak technology base. Entrepreneurs and government have became more concerned with short term measures , consequently the interest in nuclear energy , a long term endeavor, has taken a long step back. While the plans to pursue renewable energies continued unabated, the import of electric energy and oil and gas has increased in the recent two years. New licenses have been recently awarded for eight gas power plants. Prospects on whether the increase of energy demand will continue in the face of the ongoing economic downturn will be analyzed in the paper. Elections planned during 2009 for the European parliament, local bodies and the national parliament are not expect to change the government plans for energy. The ruling party may loose its absolute majority , which will oblige it to make alliances with the left that also not inclined to promote nuclear energy.

4. **Plans and implementation of renewable energies in Portugal.** An update will be provided on how these plans have been implemented a as well as an indication of prospects.

5. **The Portuguese Society of Physics.** The Portuguese society of Physics took an active role in this year session of the European Physics Society. The session prepared a position paper on the future of the nuclear option, which was adopted by all European members of the Society. The paper concluded that:” No one source will be able to fill the need of future generations for energy. The nuclear option, incorporating recent

major advances in technology and safety, should serve as one of the main components of future energy supply. There is a clear need for long-term research, development and demonstration programmes as well as basic research into both nuclear fission and fusion and methods of waste incineration, transmutation and storage. Ways must be found to inform the general public on how to assess relative risks rationally. Everybody participating in the decision making process needs to be well informed about energy issues. It is an important task of European science and research to ensure this."

6. **The way forward**. Based on the position above, the Portuguese Society of Physics plans to establish a National Association for the promotion of nuclear energy in Portugal and prepared detailed bylaws. This association will be open to the business and academic communities, media and other interested parties to initiate a public debate on nuclear energy. This proposal is in line with the recommendations made in our 2008 paper. The constitution of this association has not yet met much success in attracting members either from the business community be them users and providers of energy. However efforts continue.

Challenges and Opportunities to Launch New Nuclear Power Programmes

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The large majority of the current fleet of nuclear power plants are water-cooled reactors. Their efficient and safe operation is a key factor in assuring that nuclear power will meet both the current and the future energy needs. Refurbishment and life extension, as well as power up-rating of existing plants will be a major activity.

In order to promote the peaceful utilization of nuclear energy, in 1970s, China made a decision to develop nuclear power and further determined the technical direction of water cooled nuclear power plants. After more than 30 years' effort, in December 1991, the self-designed water cooled nuclear power plant-Qinshan Nuclear Power Plant was established and started power generation, which ended the history of no nuclear power plant on China mainland and achieved a new breakthrough. Since then on, the nuclear power plant in our country gradually developed and achieved certain accomplishment. However, if comparing it with the advanced nuclear power technology in the world, the development nuclear power in China still lagged to some extent.

1. Actuality of water cooled nuclear power plants

Up till now, there are totally 11 sets of nuclear power generators under operation, all of which are water cooled nuclear power plant. The Installed Capacity is 9124 Megawatt. In 2008, the nuclear power generation reached 69.22 billion KW/H, occupying 1.3% of the total power generation in China.

2. Opportunities for Water Cooled Nuclear Power Plants

2.1. During the occurrence of present financial crisis, the government has increased the input in infrastructure facilities and adjusted the structure of energy. It also proposed to develop nuclear power energetically. In accordance with the Plan on Sustainable and Long-term Development of Nuclear Power, by 2020 in our country, the installed capacity of nuclear power under operation will reach 40 million KW and the nuclear power capacity under construction will be up to 18 million KW. At present, the plan is still being revised and the final numbers may even exceed the current ones.

2.2. Presently the operation performance of water cooled nuclear power plant in China is fair good. In 2008, 4 of the 7 power units in our company were listed as advanced level in the world nuclear power industry, which lays a foundation for better development of water cooled nuclear power plan in China.

2.3. The projects under construction are carrying on smoothly. Currently, totally there are 18.92 million KW of water cooled nuclear power plan units are under construction along with many units with permit of implementation and potential plant addresses. The first AP1000 in the world is also in process now.

2.4. The core competitive strength of self-innovation was improved. In our country, the engineers are capable to design 2nd -generation improved water cooled nuclear power plant with million KW nuclear power unit. The design is characterized in “allocation of single reactor” and “117 reactor core”, which has been recognized by all the domestic experts. At the same time, based on the constant improvement of 3rd –generation nuclear power technology.

2.5. The degree of self-determination was reinforced, the equipments are gradually manufactured domestically, which helps to decrease the cost efficiently. To build a nuclear power plant, the investment in equipment may take about 45% of the total investment. As the equipments are gradually manufactured domestically, the cost for building a nuclear power plant is deducted greatly. The rate of domestically made equipment among all equipments in the project of Qinshan Ⅱ is 55%, increased to 77% in the expansion project of Qinshan Ⅱ. In other projects like Fuqing, Fangjiashan, the rate of whole domestically made equipments will reach 75%. In Sanmen project, the rate of conventional island and BOP will also reach 50%.

2.6. EPC(Engineering Procurement and Construction) mode was adopted in the sub-contract of nuclear power plant to finish the construction of plant. Such mode was initially used in Fuqing and Fangjiashan Project. CNPE is the only subcontract company of nuclear power plant led by design with the most powerful strength, and the most complete equipments.

2.7. China also attaches importance to the export of nuclear power and actively promotes the export work of K1000 project to Pakistan by organizing corresponding groups of people.

3. Challenges for Water Cooled Nuclear Power Plants

3.1. Although the 3rd generation nuclear power technology(AP1000) is developed and advanced, there is no whole-process check and inspection from construction to operation in the world. Neither the equipments owners nor the suppliers are capable to provide any valuable experience for reference. Therefore, the model project of the 3rd generation nuclear power technology is inevitably at certain risk.

3.2. Nuclear power entered the construction period of large scale. Therefore, how to optimize the project management, integrate group resources and improve construction efficiency become the main problems of nuclear power plant construction.

3.3. As the construction of nuclear power entered the period of large scale construction, nuclear power market gradually raises higher and higher demand on the capability of nuclear power equipment and its technology.

3.4. The construction of nuclear power entered the period of large scale construction. With the increase of newly established nuclear power plants, more professional staff is required to be trained. At the same time, the flow of work staff may weaken and reduce the experience and skills of nuclear power plant operation, which brings large negative influence on the operation of nuclear power station. Therefore, the human resource in nuclear power industry needs to be managed reasonably and thoroughly.

3.5. How to deduce the cost is also one of the challenges for the future development of water cooled nuclear power plant. The project of Qinshan Ⅱ, which is self-designed by Chinese, costs 1330 USD/KW, while the other construction of nuclear power plant at the same time costs 2000 USD/KW as the equipments and technology are introduced from foreign countries.

3.6. Challenges raised by constant increase of nuclear power performance. WANO has put forward the Six Major Lessons for the operation of nuclear power in 1990s in combination with the assessment experience.

3.7. Risks of nuclear power under financial crisis. Under the background that the world economy is fluctuated greatly, risks of investment control and cost control caused by changes of exchange rate in the introduction and export of nuclear power project and import & export of equipment also exist.

3.8. The loss of experience on nuclear power and the urgent demand of knowledge management.

To conclude, the water cooled nuclear power plant in the 21st century faces both opportunities and challenges. But we are confident that water cooled nuclear power plan will illuminate the future of China soon.

Expansion of Nuclear Power in Mexico

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Mexico has a Nuclear Power Plant located at Laguna Verde Veracruz. It has two General Electric BWR reactors of 680 MWe each, The first reactor started operating in 1990 and the second one in 1995. Nuclear power contributes about 4.4 % of the total electricity produced in México. The Mexican regulatory body has given the two reactors an operation license of 30 years.

The Laguna Verde Nuclear Plant was uprated to 5% of its nominal capacity in 1999 and at the moment there is an ongoing project to uprate it to 20%, this however, implies changing major components such as the turbines.

At the present time there are not government official plans to built new NPPs in Mexico; however studies about de technical and economical feasibility new nuclear capacity are being carried out by the CFE and the Mexican Nuclear Research Institute.

In México the production of nuclear power is reserved to the state trough Federal Electricity Commission (CFE). The Program of Works and Investment of the Electrical Sector (POISE) is an annual national electricity program prepared by CFE taking into account the electricity needs for the following 10 years.

The Mexican government has stated in the POISE 2009-2018 plans to install close to 18,000 MWe by the year 2018 of new electricity capacity, of this amount 15,500 MWe will be produced using conventional sources and 2,500 MWe will be defined in the future considering technologies that can include nuclear power.

The most viable option to increase nuclear capacity in Mexico besides the current reactors uprate program is to build new reactors in the same site of Laguna Verde. This site was originally planed to house up to 4 reactors.

Another very important issue is the operational license extension to at least 60 years, this can only be archived through a life extension program which is currently underway. The Mexican Nuclear Research Institute has been carrying out a materials surveillance program to determine the current state of the reactor vessel as well as the internals and cables, the results of this program will be the basis for a license extension from the regulatory commission.

It is necessary to carry out a more thorough investigation on the cost of nuclear power, specially in the current world economic crisis where there are huge variations in costs an great uncertainties in the estimation of future energy costs.

Mexico´s oil production is on the decline and is necessary to find alternative energy sources to secure a bigger energy independence through economical and environmentaly friendly options.

Promote the CPI Nuclear Power Development to Boost the Self-reliance in China

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Nuclear power renewal promises to energize electricity generation worldwide and help address concerns about greenhouse gas emissions. The strong safety performance of nuclear power in China, the growing demand for energy, and the increasing awareness of the environmental benefits of clean nuclear energy, form the foundation for accelerating the nuclear power development that can support China energy security, economic prosperity, and environmental quality goals in the 21st century. On October 2007, the National Development and Reform Commission of China issued the medium- and long-term national plan (2005-2020) on nuclear power development. China is aiming to have a nuclear power operating capacity of 40 GW by 2020, accounting for 4 percent of the nation's total power capacity. Recently, the Chinese government is pondering to increase further the development of nuclear power in the country.

China Power Investment Corporation (CPI) is one of the three authorized corporate utilities to develop, construct and operate the nuclear power plants as the controlling shareholder. Following the national energy policy, CPI has established the corporate strategy and the program on the nuclear power development. CPI is planning to take the 20 percentage of the total domestic nuclear power capacity by 2020.

With the orientation of the CPI nuclear power development strategy and the program, the internal and external business environment and resourses were analyzed in detail for the CPI nuclear power development. Furthermore, the challenges and the opportunities to launch the nuclear power development program of CPI were sorted out and described. Encountered the challenges and the opportunities, CPI has studied and stipulated the executing tactics including the management organization, the engineering construction management infrastructure, the prior adopting nuclear power technology, the human resourse and financing issues.

With these considerations, CPI decided to adopt the advanced PWR AP1000 technology as priority. CPI is building the Shandong Haiyang nuclear power project as the one of the two FOAK AP1000 projects. So far, Haiyang project is moving forward smoothly as scheduled. In addition, CPI is preparing additional new nuclear power projects in some other provinces besides of Shandong Haiyang NPP.

As one of the top 54 largest state-owned corporations, CPI is endeavoring and responsible to boost the self-reliance on the nuclear power developmemt in China through CPI's efforts as well as the comprehensive and sound cooperation with the nuclear power entities worldwide.

An Overview of Egypt's Human Resources Strategy for the Nuclear Power Program

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Because of the limited fossil fuel energy resources and the almost fully utilized hydro energy, Egypt has been considering for sometime the introduction of nuclear energy for electric power generation. Egypt initiated a nuclear energy program in the 1970s but it came to a halt following the Chernobyl accident in 1986. Since that time the government has commissioned several feasibility studies, which supported the case for embarking on the nuclear plan. Egypt's electricity demand has been increasing by almost 10% per year over the past decade, and power consumption is likely to continue to increase rapidly. In September 2006 the Egyptian government announced that it is considering the nuclear energy as an option for electricity generation because of the improvement in nuclear safety and the pressures on the energy sector due to the sharp increase in oil prices. In Oct 2007 President Mubarak announced that Egypt is to build several nuclear power stations, which represents strong national commitment to the nuclear power programme. Following that the government will develop the program in cooperation with the International Atomic Energy Agency (IAEA) "within a framework of transparency and respect of commitments to the nuclear non-proliferation system."The current activities includes establishing the master plan of the nuclear program based on Milestones in the Development of a National Infrastructure for Nuclear Power, the concern now is directed to Phase 2: Preparatory work for the construction of a nuclear power plant after a policy decision has been taken, which is linked to Milestone 2 - Ready to invite bids for the first nuclear power plant, for which the Human Resources Management plays a key role. The objective of this paper is to highlight human resources management policy**,** workforce planning as well as training requirements and related considerations for the nuclear power plant project in Egypt.

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Regulation and License for NPP in Indonesia

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Any activity relating to nuclear energy shall maintain safety, security, safeguards, health worker & public as well as environmental protection, according to article 16(1) act No.10 year 1997 on Nuclear Energy.

The act No.10 year 1997 on Nuclear Energy stipulated independent government agency as nuclear energy regulatory called BAPETEN (Nuclear Energy Regulatory Agency). BAPETEN has task to make regulation, licensing process and inspection. Bapeten shall have a licensing and inspection system in fulfilling his function to issued licensing according to Nuclear Energy Act.

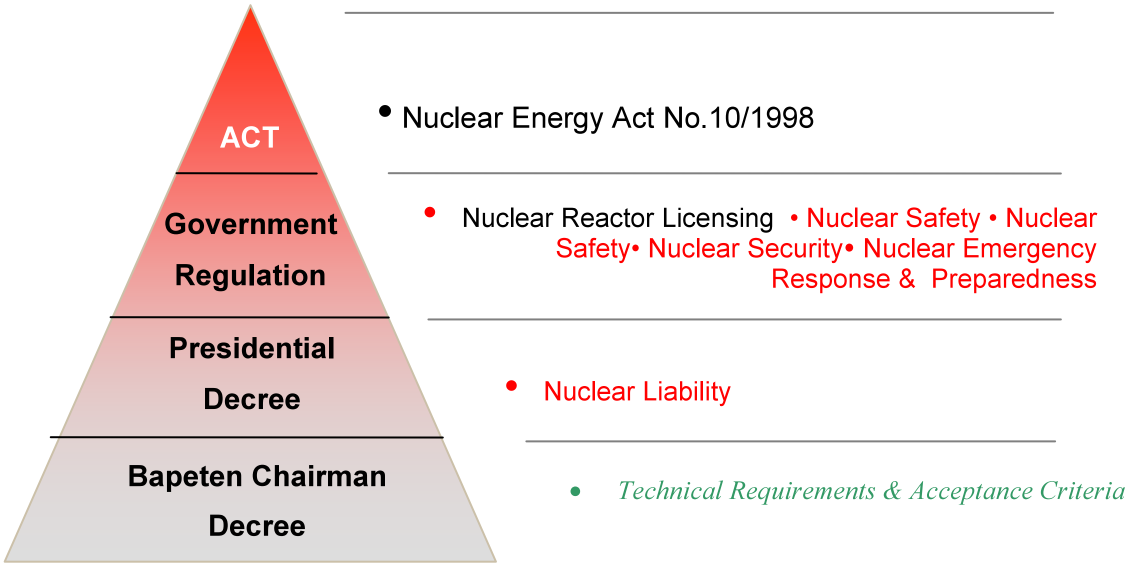
In Article 17 stipulated that any nuclear energy utilization or construction and operation of nuclear reactor and other nuclear installation as well as decommissioning of nuclear reactor shall have license.

Requirements and process to issue license for nuclear reactor stipulated further in Government Regulation no. 43 year 2006.

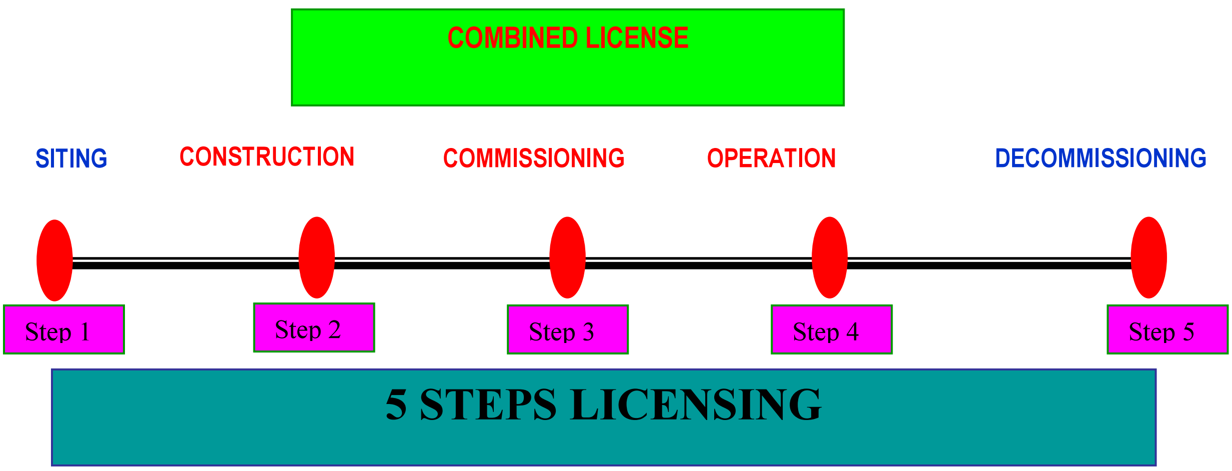
The purpose of the Government Regulation No. 43 is to stipulate licensing of nuclear reactor in order to assure the health of worker and public, environment protection and security of nuclear material and nuclear facility.

Requirement and guidance in licensing NPP mandated by Government Regulation or Presidential Decree stipulated in Bapeten Chairman Regulation

National Nuclear Legislation Hierarchy



NOTE: *red letter represent documents to be developed and established by the end of 2009 & green letter established by end of 2010*



A Nuclear Power Plant for a Developing Country: The Best Option

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In an International Conference on Topical Issues in Nuclear Installation Safety held last year in Mumbai, India I presented a paper entitled "Launching a Nuclear Power Programme – a third world country's perspective" - IAEA-CN-158/9. [1]. In it I identified some real constraints encountered by a developing country while trying to introduce a nuclear power programme. This were inadequate base infrastructure, financial incapability and lack of skilled manpower.

However a topic of similar importance if not more is the question of the best type of reactor a low GDP developing country should go for were it to make a definite decision to generate nuclear power.

Granted each developing country has different needs depending on its economy, national requirements, grid infrastructure, interregional planning and alternative resources. For example Kenya has an electricity per capital consumption of 145.803 kWh per capita which is increasing dramatically.[2] But were a real decision made to deploy, considerations on the best and appropiate type of reactor should be evaluated. Considerations would be on the reactor scalability; that is moving from a small reactor design to scalable medium reactor as power needs and grid capacity are matched.

To developing countries a myraid of opportunities for which reactors to go for is available with current, future and developing technologies presenting innovative designs such as pressurized water reactors, boiling water reactors, pebble bed reactors and advanced heavy water reactors.

Water cooled reactors are increasingly being viewed as maybe the best options for developing countries considering the many potential avenues for innovative designs they have.Chief among this may be their superior safety from primary system component failure that is provided by a series of safeguard mechanisms. This is particularly appealing to a developing country considering the enormous loss that would be occassioned if a component failure in the primary system spread into other secondary systems.

However there is the possibility of potentially high maintenance costs associated with the seemingly inaccessibility of primary components were primary failure to occur.

Despite this water cooled nuclear reactors remain as the best way into the future for developing countries that intend to deploy a reactor.

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Human Resources Development for Thai Regulators

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Thailand has a plan to operate two 1000MW Nuclear Power-Plants (NPPs) in 2020 and 2021 respectively. As for the preparation, tasks are divided into 2 major categories: revision of nuclear legislations in Thailand, and HRD.

Revision of National Nuclear Legislations

It is necessary to revise Atomic Energy for Peace Act and related legislations that had become outdated. The aim is to empower the Act to have authority and responsibility called for in many International Standards. For this purpose, we have been studying treaties and contracts between countries involving with NPPs; in order to assess agreements or legislations that they are fully covered important aspect from every treaty.

Human Resource Development

Office of Atoms for Peace (OAP), as the nuclear safety regulator of Thailand, foresees the necessity of Human Resources Development (HRD), as a key for successful preparation of NPPs safety regulations. Since Thailand does not have any clear and continuous policy for application use of nuclear energy, we are currently facing the personnel shortage, both in term of number and quality. Moreover, BNSR used to be responsible for both R&D and regulation, which resulting in an inadequate focus on regulation. This also results in very limited affords on recruitment of specialists in regulation field.

Management

At the national level, the situation of global energy resource and the rising concern about climate change have given The Ministry of Energy, which is responsible in searching energy resources for our country, reasons to consider using NPPs. As of now, Thailand Electricity Generation Authority (EGAT) is currently searching for suitable sites for NPPs. This project's reports are scheduled to be submitted to Thai government in the early of 2010 to decide if the NPP project should continue. These reports contain preliminary feasibility study of NPP project, including suitable sites' location, which will be evaluated by BNSR later. Meanwhile, BNSR will be working on HRD, via 2 main projects

1. Personnel performance assessment projects, specialized in nuclear safety regulation. The first phase is a process of acquiring knowledge qualification and characteristic of nuclear safety regulation human resources. This phase focuses on analyzing differences in performances among human resources, for the purpose of designing an effective curriculum that will narrow down differences in performances among human resources. The next phase

will focus on defining courses for operating human resources in nuclear safety regulation department.

2. Enhancing human resources in nuclear safety regulation project. This project is an intensive program to accelerate personnel's capability in performing specialized tasks. In the fiscal year for 2009, the project will be focused on increasing potentiality for analyzing and evaluating safety issues of NPPs' sites.

Conclusion

Despite the efforts, there are many difficulties presented in the following aspects.

1. Public policy is unclear, resulting in limited budget.

2. Development processes take time, since both training and accumulating experience cannot be done within a short period of time.

3. There is inadequate in human resources.

Future needs

Hence, BNSR is in need of the following assistances from IAEA as soon as possible, in order to be ready in time for NPPs' project, as mentioned above.

1. Strategic planning for development in human resources

2 .Intensive training programs, Mentoring Program or "on the job" instruction.

3. Technical Knowledge on standards and criterions of site evaluation

4. Peer evaluation in order to reassure that development is on the right track.

Localization of Manufacturing Capabilities in Setting Up Nuclear Power Plants

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Nuclear renaissance is now imminent and is inevitable in view of the global warming concerns .Indian government in its integrated energy policy document has planed for raising nuclear power capacity to generate 63 GWe by 2030. This envisages estimated investments of US $22 billion in next 15 to 20 years.

In the light of this, for setting up Nuclear Power Plants, it becomes very important to ensure supply-chain for materials and components and putting in place cost effective project management to complete the projects on time and within the budgets.

In this context, the participation of industries and their preparedness to meet the challenges are necessary. This would also require investments towards up gradation of manufacturing technology, training of manpower and mobilization of resources at the construction site .The industry would also need to enhance detailing and design engineering capabilities for the plants.

In this paper, various issues with regard to project cost, regulatory and licensing, technology and gestation period etc for new build plants relevant to manufacturing industry are discussed. The plans for enhancing manufacturing capabilities for the critical path items of the project schedule with viable business, ensuring returns to stakeholders and financing and investment cycle are brought out.

The various steps and initiatives being taken by Bharat Forge Ltd a flagship company of Kalyani group for supply of forgings for nuclear reactor, vessels, steam generators , turbine generators and other safety critical components including pumps, valves, pipes and tubes , and other integration work for the balance of plant are summarized.

The company has planned significant investments for setting up an integrated manufacturing facility to clear the global bottleneck in the supply of ultra-heavy forgings for light water reactors.

Civil structuring involves execution by prior qualification with proven skills for specialized civil works needed in the construction of nuclear plants. Another group company “Kalyani Technical Management Services [KTMS] is also gearing up to enhance its capabilities to undertake construction projects and execute EPC contracts.

In the context of Indian Nuclear Power Program, the various initiatives taken for overall human resource development to meet the demands requiring skills in high end technology manufacturing and project management are included.

ACR-1000 - Designed for Constructability

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One of the key aspects to be considered in the delivery of a Nuclear Power Plant is the security of the construction schedule and the need for lower construction costs. Many industries are using skids, modules and prefabrications to enhance construction productivity, reduce schedules and thus reduce costs. The leaders in this regard have traditionally been in the off-shore oil and gas, chemical, refinery and ship building industries.

The concept of using modules has been utilized in Nuclear Power Plant design and construction. Atomic Energy of Canada Limited (AECL) has had considerable success at the Qinshan Nuclear Power project in China with the use of modularization, which proved extremely effective in the ability to organize parallel construction activities and shortening the schedule. Extensive use has been made of skids and modules in Japan and this also has proven effective in shortening schedules in the construction of nuclear power plants. Secondary benefits of modularization and prefabrication include decreased site congestion and logistical issues, increased worker safety and better quality control of fabrication. Modules and prefabrication allow work to be shifted to areas where skilled trades are more readily available from a site where skilled trades are very limited.

One of the objectives of the ACR-l000 project is to produce a design that allows for a very secure construction schedule. The construction method and strategy, consisting of extensive use of prefabrication and modularization was defined very early in the ACR- 1000 conceptual phase of the layout and design process. This has been achieved through a constructability programme that integrates the civil design with site erection and module installation. This approach takes the concept of modularization to an entirely new level, in which the use of modules is built into the design from the start, rather than backfitting modular construction into a conventionally designed plant.

This paper presents the ACR-l000 construction strategy and methods and shows examples of how the integrated civil design approach with modularizalion and prefabrication is utilized to shorten the construction schedule and reduce the project risk.

The following aspects are described in the paper:

— The use of very heavy lift (VHL) cranes has made it possible to leave the top of the containment structure and install heavy items through the ‘open top” of the reactor building.

— Scheduling of parallel construction activities is achieved through the use of modules and prefabrication in the construction schedule.

— Shortening the schedule is possible through the use of modern construction technologies, such as prefabricated rebar, large volume concrete pours, prefabricated permanent formwork, skid mounted modules and pre-fabricated pipe assemblies.

— The ACR-l000 plant layout has been designed for constructability, incorporating vertical installation compartments, which in turn are divided into construction volumes. The construction volumes represent volumes of the plant that will be issued as complete work packages during construction. Major room modules, where implemented, may consist of a floor, the material suspended below the floor and the equipment and material mounted on the floor.

The use of prefabricated permanent formwork (PPF) for the floors. A typical PPF would consist of the steel liner plate serving as the formwork, the truss elements, bottom layers of reinforcement, the penetrations and embedded parts in their final position. Floor slab construction work proceeds with connection of the bottom rebar layer to the exposed wall rebar using mechanical couplets, followed by the top layer of reinforcement installation and connection of the top rebar to the exposed wall rebar, also with mechanical couplers. Concrete is then poured and cured, and the PPF becomes part of the structure left in place. The main advarnage of this structural concept is that the PPF is a pre-engineered job and the PPF can be lifted into place by the VHL.

— The utilization of Laser Measurement and Spatial Analysis/Modelling for the manufacturing of modules to a precise dimensions so that installation and interconnections can be accomplished in the field with the minimum amount of labour and time to further accelerate the completion of the Nuclear Power Plant.

— The design considerations that are used in sub-dividing the equipment, piping, supports and civil structural members into suitable modular volumes. The ACR approach to modularization is a departure from convention in that it *designs* the systems and structures to be modularized as opposed to dividing *pre-designed* systems into modules or conventional stick-built.

— The benefits of using this construction approach in the overall construction schedule for the ACR-l000 is finally discussed.

The Protection of Containers for Fresh and Spent Fuel at Extremal Transportation Operating Modes in and Around a Nuclear Reactor's Portal

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The certified containers for fresh and spend fuel (further the containers) are able to survive dropping from a distance of 9 meters on a concrete table top [1, 2].

In present-day nuclear power plants the hoisting height of the containers at cargo handling is up to 40 meters. Possible fall of the containers on a concrete table top will probably result in its destruction.

The goal of the paper is the substantiation of a new shock-absorbing gravel-sand cushion that is designed to be installed under a Nuclear Reactor's Portal. Furthermore, there are determined the physical and mechanical specifications and the dimensions of the new shock-absorbing gravelsand cushion. A new construction of shock-absorbing gravel-sand cushion must absorb the kinetic energy of a container so that the unabsorbed kinetic energy would not exceed the value corresponding to its fall from a distance of 9 meters on a concrete table top.

The shock-absorbing cushion mounting (composed of asphalt, gravel and sand) under portal crane of a Reactor Building will insure structural strength and impermeability of the containers in case of its fall from the maximum possible height at all transportation operating modes.

At the beginning of the paper the possible trajectories, positions and kinematical parameters of the containers are determined from a fall from the distance of up to 40 meters. These calculations are made for the cases of cranage failures.

In a 3-D space, by means of FEM contact problems of the container and the shock-absorbing cushion interaction are solved. The kinematical parameters determined earlier are used as an initial condition.

It was shown that the overload coefficients for the containers do not exceed the admissible values at possible impacts of the containers with the shock-absorbing cushion.

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Development of Combined Modularization Technology for APR+ (Advanced Power Reactor) in Korea

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KHNP is currently operating 20 nuclear power plants and constructing 8 units in Korea. With the accumulated experience of operation and construction of NPPs over the past 50 years, KHNP tries to develop the original nuclear power reactor (APR+). The R&D project for 'combined modularization', which is a part of APR+ common core technology development, is being carried out and will be implemented by July 2010.

This R&D project was initiated from the feasibility study on the modularization of structure and system facility for NPPs (2002). On the basis of the feasibility study result, The SC (Steel plate Concrete) structure technology development had been performed from 2005 to 2008, and the SC structure technology is under audit by the regulatory institution prior to the application for nuclear construction project.

The research of combined modularization is an extension from the SC structure technology development. The research is divided into three phases. During first phase, by July 2010, the applicability of SC structure enlargement will be reviewed and conceptual design of combined/equipment modularization will be performed. In second phase, the basic design and construction feasibility evaluation of the screened combined/equipment modularization will be completed within July 2013. Afterward, in the final phase, optimized modularization technique will be realized through technical supplementation by Dec. 2015.

Two main objectives will be achieved in the first phase study. The first one is to obtain design reliabilities for the SC structure by evaluating its applicability in auxiliary/compound building. The other one is to establish modularization requirements for design, procurement, manufacture and installation. To ensure these objectives, the flow sheet of technology development is greatly divided into 5 steps as follows.

First step is to establish the screen criteria for adopting modularization. It's possible to catch the attitude of impact factor of modularization and evaluate the adoption of module's screen criteria using systematic analysis method.

Second step is to select the modular object using the former research result and reference plant (APR1400). The modularization object is chosen and then these potential modules will be evaluated in views of design, procurement, manufacture and construction.

Third step is to perform the example design for each module types practically. Using module list, representative module is selected each module types and example design is performed by design data of reference plant. Through such process, system and manufacture design requirements for modularization are established and standardized procurement specifications are prepared.

Fourth step is to evaluate constructability and analysis economical efficiency. The constructability by the design and mockup test result is evaluated and the guideline is prepared. Also economical efficiency by cost analysis is evaluated and construction cost according to combined modularization is quantitatively analyzed.

Fifth step is to establish the modularization strategy. The development technology is incorporated to nuclear construction project step by step and feedback between consecutive research development and project adoption is harmonized.

The main activities during first phase of combined modularization are as follows.

First activities are the evaluation of applicability of SC structure and establishment of design technology. These activities include the development of guideline for SC structure joint, selection of structure member where SC structure could be applied.

Second activities contain design and construction technology for combined module. The adoption targets for combined module based on SC structure are selected according to schedule and design requirement. On process of design, manufacture and construction for mockup of combined module, the abilities for manufacture and construct are evaluated and the construction guidelines are prepared.

Third activities include technology development of design and manufacture for equipment module. The concept and basic design about the selected equipment module is performed and the design and manufacture methodology for equipment module is established.

Fourth activities contain establishment of shorting construction schedule for adopting SC structure, combined module, equipment module in nuclear projects and economical efficiency evaluation by cost analysis about the variously developed module.

Fifth activities consist of establishment of screen criteria and evaluation of the selected module. After founding the screen criteria, the facilitating and impediment factors for modularization are identified and the enlargement plans for the modularization are prepared.

Modularization method for nuclear construction is innovative and powerful tool to shorten the construction schedule and improve the construction productivity and over-jump the current design, procurement and construction system. Therefore KHNP will establish the design, manufacture, procurement and construction technology about modularization method throughout the first period of APR+ combined modularization technology development. Also the modularization technologies will be incorporated in nuclear construction projects and will be continuously improved during the second period.

Development of Coupled Dynamics Model for VVER Reactors

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As a key of core knowledge for the design of various types of nuclear reactors and operating experience of more than thousand reactor-years, the discipline of reactor physics has been advanced continually in the past six decades and had led to development of many core dynamic analysis methods and computer codes. The present age computer resources have permitted the reactor safety analysis of large reactors to be carried out with detailed thermal hydraulic coupled 3D neutronics models to get the so called 'best estimate' results. Such detailed multiphysics modeling procedures minimize computational uncertainties. The operational flexibility of nuclear plants can also be improved by utilizing accurate 3D coupled neutronics/thermal—hydraulics calculations for safety margins evaluations.

The present paper describes a brief review of evolution of reactor kinetics analysis methods from simple point kinetics tools to complex multiphysics coupled code system being used today for best estimate analysis to avoid unnecessary over design scenarios in competent power industry market. It also describes the methodologies and practices involved in coupling core thermal hydraulics and neutronics calculation being followed in AERB, India.

An indigenous effort is going on in Atomic Energy Regulatory Board (AERB), Mumbai, India to develop in-house computer code systems to analyze wide spectrum of transients in large water cooled reactors. A computer code, TRIKIN has been developed in this regard to carry out detailed 3D kinetics and coupled core thermal hydraulics calculations for VVER reactors which are being constructed in Kudankulam, India. Neutronics model uses a computationally efficient algorithm based on the flux factorization technique employed by the Improved Quasistatic (IQS) method. The space and time dependent group fluxes are factored into space and time dependent shape functions which change slowly in time and a rapidly changing amplitude function which is only a function of time. The solution is obtained for these functions on two level time axis, with amplitude function solved on very finer time steps, whereas shape functions solved at larger time steps. Core thermal hydraulics is modeled by equivalent multichannel representation and mass, momentum and energy conservation equations are solved for each channel neglecting the cross flow among the various channels. The coupled code has been validated against a series of AER benchmark problems (AERDYN-l, 2, 3) for VVER-44O core. The benchmark problems are asymmetric rod ejection accident analysis with different level of complicacy in reactivity feedbacks. Presently the model capability is specific to the analysis of VVER reactor system which has triangular lattices fuel pin arrangement but the methodology can be extended to any type of nuclear reactor.

To demonstrate the capability of the TRIKIN model, few selected results of benchmark problems shown in figures. The figures show that TRIKIN results are in good agreement with benchmark results.

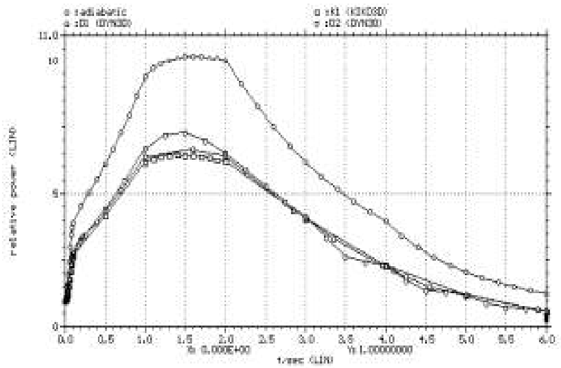

Fig 1A: Normalized Total Power During The Transient (Benchmark, AERDYN001)

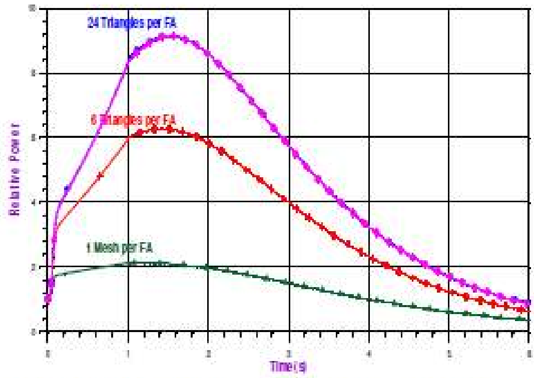

Fig 1B: Normalized Total Power During The Transient (TRIKIN)

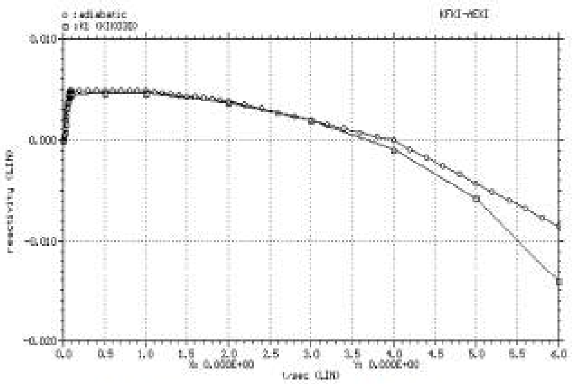

Fig 2A: Reactivity During The Transient (Benchmark, AERDYN001)

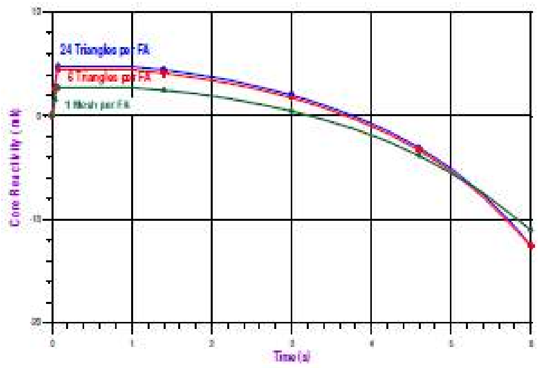

Fig 2B: Reactivity During The Transient (TRIKIN)

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CANDU®; Shortest Path to Advanced Fuel Cycles

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The global nuclear renaissance exhibiting itself in the form of new reactor build programs is rapidly gaining momentum. Many countries are seeking to expand the use of economical and carbon-free nuclear energy to meet growing electricity demand and manage global climate change challenges.

Nuclear power construction programs that are being proposed in many countries will dramatically increase the demand on uranium resources. The projected life-long uranium consumption rates for these reactors will surpass confirmed uranium reserves. Therefore, securing sufficient uranium resources and taking corresponding measures to ensure the availability of long-term and stable fuel resources for these nuclear power plants is a fundamental requirement for business success. Increasing the utilization of existing uranium fuel resources and implementing the use of alternate fuels in CANDU® reactors is an important element to meet this challenge.

The CANDU® heavy water reactor has unequalled flexibility for using a variety of fuels, such as Natural Uranium (NU), Low Enriched Uranium (LEU), Recycled Uranium (RU), Mixed Oxide (MOX), and thorium. This CANDU® feature has not been used to date simply due to lack of commercial drivers. The capability is anchored around a versatile pressure tube design, simple fuel bundle, on-power refuelling, and high neutron economy of the CANDU® concept. Atomic Energy of Canada Limited (AECL) has carried out theoretical and experimental investigations on various advanced fuel cycles, including thorium, over many years. Two fuels are selected as the subject of this paper: Natural Uranium Equivalent (NUE) and thorium.

NUE fuel is developed by combining RU and depleted uranium (DU) in such a manner that the resulting NUE fuel is neutronically equivalent to NU fuel. RU is recovered from reprocessed light water reactor (LWR) fuel and has a nominal ^{235}U concentration of approximately 0.9 wt%. This concentration is higher than NU used in CANDU® reactors, which is 0.71 wt%. Therefore, the RU is blended with the DU in such a manner that the NUE has a ^{235}U concentration similar to NU. NUE fuel in a CANDU® reactor has been shown to be technically feasible. This fuel offers the simplest, quickest and lowest cost path to efficient utilization of alternative fuel sources in the currently operational CANDU® reactors without modifications to their existing licensing basis.

The use of NUE in CANDU® reactors addresses the challenge of dealing with spent LWR fuel that would otherwise require monitored storage or costly re-enrichment and handling of highly radioactive fuels for reuse in LWRs. Using RU in existing nuclear power reactors will improve the utilization rate of NU resources and ultimately improve the sustainability of fuel resources. In addition, NUE reintroduces the fissile content in DU back into the fuel cycle; an effective application of an otherwise limited-use by-product from the uranium enrichment process.

There is a wealth of global experience with RU, however, due to previously low economic incentives in LWRs, its utilization in nuclear reactors has been limited. With increased global requirements for uranium resources and fluctuating uranium prices. RU use in CANDU® will play a vital role in the nuclear renaissance. Currently AECL and its partners in China, led by Third Qinshan Nuclear Power Company (TQNPC), has undertaken the commercial demonstration of NUE in Qinshan 3 CANDU® units. The objective is eventual full core implementation of NUE fuel at a time desired by China.

Thorium is available in large quantities in countries such as China and India, both of which have rapidly expanding economies requiring additional power sources driven by sustainable and easily accessible fuel sources. CANDU®'s excellent neutron economy, resulting from the use of heavy water as both moderator and coolant, provides for the efficient use of an LEU or plutonium driver in thorium fuel cycles. The proven CANDU® reactor is able to operate with a full core of thorium fuel without requiring major changes to the existing safety and control systems. AECL has conducted work on thorium fuel cycles, which demonstrated that both plutonium and LEU could be used as driver fuels for thorium.

The once-through fuel cycle is the simplest thorium fuel cycle that can be rapidly implemented in existing CANDU® reactors. Currently, detailed studies and accompanying design work are in progress to ensure initiation of the thorium cycle with the use of an LEU driver in order to rapidly implement its commercial application. however, maximum thorium utilization is achieved through recycling ^{233}U in a closed-fuel cycle. Over time and through evolutionary modifications, the reactor will be optimized to operate with thorium in a recycling mode.

In AECL's fuel-cycle vision, CANDU® reactors will operate in conjunction with other reactor types and use advanced fuels to produce more energy and ensure the most efficient and least costly method of utilizing LWR spent fuel recycled products. With this goal, CANDU®s will be a strong partner in ensuring the availability of long-term stable resources for nuclear power plants.

Reactor Plant WWER-1200 for the Units of AES-2006

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Federal Purpose-Orietned Program "Development of Nuclear Industrial Complex of Russia, for 2007- 2010 and prospects to 2015" provides for construction and commissioning of new standard commercial NPP Units with the advanced RP of WWER-1200 type being under development in OKB "GIDROPRESS".

Development of WWER-1200 RP is based on the standard reactor plant WWER-1000 (V-320 type) operated at 21 NPP Units in Russia and abroad.

Design bases of RP WWER-1200 are based on WWER technology (designs V-320, V-392, V-412, V-428) with its further improvement.

Among the design bases three main groups can be picked out:

- Inherent safety;
- Passive safety;
- Integrity of safety barriers.

Basic targets

— Nominal electric power,	1200 MW;
— Design service life of NPP main equipment,	60 years;
— Availability factor averaged over the whole NPP service life,	92%;
— Efficient number of nominal power hours,	8400 h/year;
— Duration of overhaul life,	8 years;
— Maximum fuel burnup over FAs	70 MW-day/kg U;
— Fuel cycle length	up to 24 months;
— Requirements for Unit load follow characteristics on the whole	according to EUR;
— Reaching the safe shutdown state for any anticipated operational occurrences, for design basis accidents and beyond design basis accidents (except for severe accidents)	within 24 hours;
— This time period is allowed to be increased up to 72 hours for design basis accident and beyond design basis accident;	
— Feed water inventory at Unit shall be sufficient for decay heat removal	within 24 hours;
— Total frequency of the core degradation	less then 10-6 / reactor-year;

- Reliability targets:
  - frequency of trips - not more than 1 per a year of operation
  - average unavailability within the design service life of a Unit - less than 1,4 % (less than 5 days per a year)

Reactor plant description, basic design solutions for RP main equipment (Advanced reactor, Fuel assembly, Steam generator, Reactor coolant pump set) and safety systems configuration (including the inherent safety principle) are presented.

The short description of further WWER reactor plant variants are presented, including:

— two loop WWER-600 reactor plant based on WWER-1200 equipment and smaller reactor with 109 FAs core
— two loop WWER-1200 reactor plant based on WWER-1200 reactor and steam generators and reactor coolant pumps of new design.

A short description of the concept of Generation IV RP with vessel-type light water WWER SCP reactor with supercritical coolant pressure is presented. WWER -SCP may become the basis for SUPER-WWER reactor.

Method and Result of Experiment for Support of Technical Solutions in the Field of Perfection of a Nuclear Fuel Cycle for Future PWR Reactors

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This paper represents conceptual approaches of statement and carrying out of experiments to validate functional safety of PWR reactors of the future, at acceptance of technical solutions on use of fuel rods with the increased length of a fuel column in fuel assemblies.

The paper represents main principles and criteria, which we use for quality check of technical solutions and developments in the field of perfection of a nuclear fuel cycle of PWR reactors of the future, first of all, from the point of view of a substantiation of safety of the future operation at change of fuel rod design. We explore the safety issues of operation of PWR reactors with fuel assemblies, including fuel rods with various length of a fuel column.

The paper discusses the ways of solving of the important problems of carrying out of critical facility experiments for verification of new technical solutions in the field of PWR nuclear fuel cycle improvement on the base of international standards ISO 2000:9000 and functional safety recommendations on functional safety of IEC (International Electromechanical Commission).

The package of new Federal Laws of the Russian Federation in the field of safety and licensing of activity of dangerous manufactures defines a major principle for requirements to the supplier of nuclear techniques and NPP as a whole. This principle is - for any moment of operation of NPP quantity indicators of risk should not exceed comprehensible social size of the established indicators of safety. On the other hand the second principle should be applied from operation of the equipment, systems or NPP as a whole to extraction of the greatest benefit: As much as possible long operation and full commercial use of resource and service properties of the equipment, systems and NPP as a whole. Realization of this principle assumes development and introduction of new technical solutions for a validation of guarantees of safety of the future operation of NPP or it separate components.

Solving the practical problems of a validation of safety use of fuel rods with the increased length of a fuel column in fuel assembly in nuclear reactors of the future, we should choose new strategies and programs of verification experiments on the base of the analysis of guarantees quality representation of safety, reliability, efficiency and benefit of operation of the NPP. From here there is a new problem of construction of the system of statement and carrying out of experiments in substantiations of functional safety of PWR reactors of the future. In view of world tendencies of development of production and services the decision of the problem is being carried out in the environment of constantly improved quality control system (QCS) of the processes of the whole life cycle of nuclear installations constructed under the Russian projects. The QSC bases on the principles of a quality management of international standards ISO of a series 9000:2000, namely orientation to a consumer lay;

system approach to management; the process approach for carrying out verification researches; the decision-making based on saved up experience, knowledge base and database; leadership of the head; constant improvement of QSC, etc.

Nowadays the validation of safety is based on calculation forecasts. According to calculation forecasts the distribution of power field in WWER-1000 fuel assemblies close to fuel assemblies with lengthened fuel column fuel rods is defined basically by the influence of compensatory volumes of not advanced fuel rods. Such situation arises in the beginning of a stage of introduction of new type fuel assemblies, when their amount in core is insignificant and regular fuel assemblies surround them. An experimental research of a core of PWR reactors of the future in the situation, when fuel loading will include fuel assemblies with various length of a fuel column, is a necessary condition of verification of calculation forecasts. The primary goal of such researches is a modeling of the situations arising at the beginning and the end of a cycle of introduction of advanced fuel assemblies and definition of mutual influence of different type fuel assemblies on a power field distribution.

To explore different type fuel assembly's core, the internal part (331 cells, WWER-1000 fuel assembly model) was lifted on some height in comparison with associate's fuel rods. The core view is presented on fig.1. Relative axial distribution of power field has been measured in fuel rods, located in the middle of the first sequence, surrounding WWER-1000 fuel assembly model, and on edge of some. The results of the experiment are presented on fig.2.

The lead experiments have shown, that the increase of axial distribution of a power field in fuel rods, surrounding WWER-1000 fuel assembly model, is observed. But this increase cannot affect the fuel rods operation safety of since it is insignificant and is in the field of where the power is approximately in 10 times less, than in the fuel rod center.

Fig. 1. The bottom of core (fuel rods of peripheral part are taken from sector of symmetry 60 deg.).

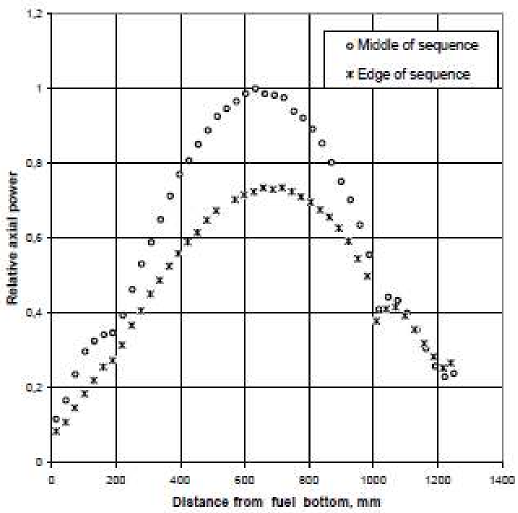

Fig. 2. Relative axial power distribution in fuel rods.

Improving Safety Provisions of Structural Design of Containment Against External Explosion

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Explosions may occur due to a variety of reasons that need to be identified and for which the probability of occurrence may need to be quantified. ACI Standard 359-07 [1] "Code for Concrete Reactor Vessels and Containments" deals with the impulse loads as time dependent loads e.g. the dynamic effects of accidental pressure P_a, the effects of pipe rupture reactions R_{rr} and Jet impingement loading R_{rj} etc. These impulse loads lie within the purview of internal explosions. However, the provisions to deal with the effects of external explosion against reactor containment are still in developmental stage. The present paper has, therefore, been directed to study the effect of external explosion on WCR containment.

A 1: 25 scale model of a typical nuclear power plant containment structure was constructed to determine the experimental relationships of airblast pressure time history as a function of surface explosion charge weight, distance to structure, structure height, as well as the simultaneous ground shock wave history. The experimental setup with explosion scenario is shown in Figure 1. The similarity of soil parameters at the experimental site with the actual site conditions was ensured in order to determine the ground shock relationship. The success of the experimental programme was dependent on the ability to accurately measure reflected pressure and to collect acceleration data from wall-mounted accelerometers.

The general practice is to utilize the air blast pressure values in the structural analysis and design against external explosion. The ground shock parameters are usually neglected during blast resistant analysis and design. In this paper, not only the airblast parameters have been studied but also the ground shock parameters have been dealt with. Therefore, the paper deals with the experimental determination of relationships of following airblast and ground shock parameters against scaled distance on containment scaled model.

Airblast Time History Parameters

(a) Peak pressure (P_{so})

(b) Shock wave front arrival time (T_a)

(c) Rising time (T_r)

(d) Decreasing time (T_d)

(e) Duration of the positive pressure phase (T)

Ground Shock Time History Parameters

(f) Peak Particle Acceleration (PPA)

(g) Arrival Time (t_a)

(h) Shock Wave Duration (t_d)

(i) Time lag between ground shock and air blast pressure arrival at structures (T_{lag})

FIG. 1: Experiment setup and explosion scenario

The ground shock wave results have been compared with that of CONWEP [2]. The variation of results is due to curved surface of containment model. It is concluded that an accurate analysis of structural response and damage of structures to a nearby external explosion requires simultaneous application of ground shock and air blast pressure time history parameters. The research work and the equations drawn may be utilized for the design of AWCR against external explosion.

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The Westinghouse AP1000 – An Advanced Passive Plant for A Safe Nuclear Future

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In the last decades the world demand for energy was growing rapidly and despite the economic crisis no basic change in this development can be expected for the future. Especially the developing countries with fast growing populations and economies will have an increasing need for energy in all forms while available resources are decreasing. The extreme volatility of energy prices in the last years has shown how dependent on cheap and reliable energy we are and how dangerous it is to focus only on few sources. On the other hand the excessive use of fossil fuel in the last century has led to a man made climate change by emission of greenhouse gases. The renaissance of nuclear power is one of the answers to these issues. Global warming cannot be avoided on a short but carbon free nuclear technology can help limiting the effects of climate change and provide billions of people with cheap and reliable energy at the same time.

However, a worldwide “nuclear renaissance” has some prerequisites. Beside economic competitiveness advanced nuclear safety based on proven, evolutionary technologies is the most important condition for the acceptance of new nuclear power plants. Half of the world’s nuclear reactors are based on proven Westinghouse design and these plants have more than 10,000 years of operating experience. The AP1000 is the result of an evolutionary development process where advanced, passive safety features have been combined with proven components from operating plants. Significant design simplifications together with the reduction of piping, cabling, pumps, valves and seismic grade building size have led to reduced investment costs. A consequent modularization enables the plant to be constructed within three years. Passive safety functions relying only on natural forces make the plant safer than any existing reactor. Because active decay heat removal systems do not have to be safety grade classified any more, construction, inspection and maintenance are less expensive as for competing designs. Extensive testing programs were performed for the new passive safety functions ensuring their reliability.

The AP1000 is the only design which received the NRC design certification for the new “one-step” licensing process according to 10 CFR 52. Up to now four orders have been placed in China and the AP1000 will be the standard design for the ambitious Chinese nuclear energy program. Construction of the first plant was started in March 2009 at the Sanmen site for which two AP1000 plants were ordered and four more are planned in the future. At the Haiyang site construction will start in autumn 2009. Two AP1000 were ordered for this site and six more are in planning. Additionally ten contracts have been signed in the US and several vendors are applying for Combined Construction and Operating License. The AP1000 also will be the workhouse for future new builds in Europe and other countries.

Integrity Evaluation for Elbows Based on TES Collapse Load

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Local wall-thinning due to flow accelerated corrosion (FAC) is a primary degradation mechanism of carbon steel piping system in nuclear power plants (NPPs). Because severe wall-thinning results in reducing load carrying capacity of the pipe components, adequate maintenance program should be applied to prevent the failures of piping systems. This program should be composed of several technical items such as the corrective selection of highly susceptible locations, the thickness measurement using appropriate non-destructive technology, the integrity evaluation on continuing serviceability, and so on [1, 2].

ASME Section III and ASME Section XI Code Case N-597 provide the design rules for pipe components and the requirements for analytical evaluation of pipe wall-thinning, respectively [3, 4]. Several studies had been conducted to develop an alternative criteria and to establish a structural integrity evaluation procedure for locally wall-thinned pipe components. However, most of these studies focused on straight pipe containing local wall-thinning despite the fact that wall-thinning mostly occur at elbow or pipe bend. Several numerical and experimental studies under bending load had evaluated the effect of local wall-thinning on the reduction in the collapse moment and fatigue resistance of elbows. However, there is only one experimental study performed under combined loads of internal pressure and bending to investigate the failure behavior of locally wall-thinned elbows [5]. In our previous study, it was noted that the wall-thinned elbows show different failure modes of buckling, ovalization, and crack in a broad point of view and these failure modes can be changed according to the direction of bending loads, the locations of wall-thinning, and the size of defects.

FIG. 1 shows the load definitions for bending to elbow, elastic to plastic load, plastic collapse load, and instability load. In this study, mechanical integrity evaluation model for wall-thinned elbows based on twice elastic slope (TES) plastic collapse load is shown in equation (1) and (2).

$$\mathbf{M_L = Safe\ Factor \times M_o(mp^2 + np + 1)} \qquad (1)$$

$$\mathbf{M_o = M_{NT}\left[(f_o - f_\infty)\exp\left(-3.0\frac{L}{D_o}\right) + f_\infty\right]} \qquad (2)$$

To develop this model, a lot of finite element analyses were performed as shown in FIG. 2 and the results were compared with the previous experiment results illustrated in FIG. 3. Finite element analyses results were used to obtain the failure behavior of the wall-thinned elbows with extended wall-thinning shapes and sizes of the experimented specimens. The analysis results were reviewed to examine the behavior of wall-thinned elbow under various loading conditions. All TES plastic loads obtained from experiments as well as finite element analysis were used to develop this mechanical integrity evaluation model. Correction factors were also applied to maintain the conservatism of the calculated TES plastic collapse loads.

This model can be used to calculate the TES plastic collapse load of wall-thinned elbows for various internal pressure, wall-thinning location, and bending direction. The calculated TES plastic collapse load can be used to confirm the actual safe margin between the endurable moment of wall-thinned elbow and the maximum moment allowed by construction code which can be calculated back from stress analysis requirements of piping systems.

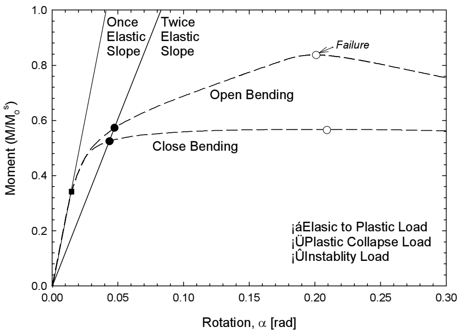

FIG. 1. Load definitions for bending moment to elbow

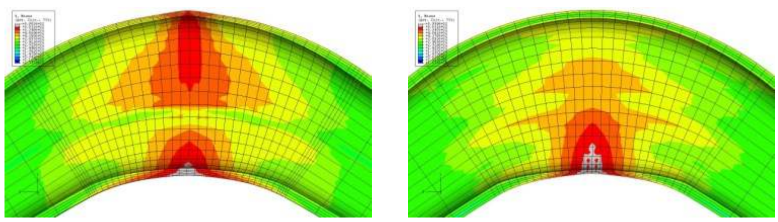

FIG. 2. Load distribution under bending moment in FEA

2θ=180, Extrados Opening moment | 2θ=90, Intrados Opening moment | 2θ=180, Extrados Closing moment | 2θ=90, Intrados Closing moment

FIG. 3. Elbows deformed in mock-up tests

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Extension and Verification of the Cross-Section Library for the VVER-1000 Surveillance Specimen Region

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The objective of this work is a generation of new version of the BGL multigroup cross-section library (47 neutron, 20 gamma groups) to extend the region of its applicability. The existing library version is problem oriented for VVER-1000 type of reactors and was generated by collapsing of thc VITAMIN-B6 problem independent cross-section fine-group (199 neutron, 42 gamma groups) library applying the VVER-1000 reactor middle plane spectrum in cylindrical geometry. The new version BGLex additionally contains cross-sections averaged on the corresponding spectra of the surveillance specimen's (SS) region for VVER-1000 type of reactors.

The extension of BGL library evolves from to the fact that the surveillance specimen of VVER-1000/320 are located on the baffle above the reactor core upper edge in a region where geometry and materials differ from these ones of the middle plane and the neutron field gradient is very high which would determine a different neutron spectrum. That is why the application of the BGL library for the neutron fluence calculation in the SS region could lead to an additional inaccuracy. In addition the discrepancy between measured and calculated foil monitors' activities exceeded 30% [1,2]. This discrepancy could be due to the uncertainty of SS geometry position as well as to the application of inappropriate cross-section library for calculation at the SS region. Comparison of the neutron spectra on the pressure vessel and at the SS position shows that the difference reaches 80%.

To assess the BGL applicability in the SS region comparative analysis of the neutron spectra were done for this region with the BGL and VITAMIN-B6 libraries and two-dimensional discrete ordinates code DORT. The difference between the spectra with energy E>0.5 MeV calculated with the libraries for the SS region is in limits of 10% except for the energy region E>10 MeV where the difference reaches 20% but its contribution to the neutron fluence with energy E>0.5 MeV is insignificant (less than 1%). The integral neutron fluence E>0.5 MeV calculated with the libraries coincides in limits of 2.3%. That means that the BGL appliance for the SS region does not lead to a significant uncertainty of the neutron fluence calculation.

Comparative analysis of the neutron spectra for diffcrent one-dimensional geometry models that could be applied for the cross-section generation using the software package SCALE, showed a high sensitivity of the results to the geometry model. That is why a neutron importance assessment was done for the SS region using the adjoint solution [3] calculated by the two-dimensional code DORT and problem-independent library VITAMIN-B6 (FIG. 1). The one-dimensional geometry model applied to the cross-section collapsing were determined by the material limits above the reactor core in axial direction z (FIG. 1) as for every material a homogenization in radial direction was done. The material homogenization in radial direction was done by material weighing Ψ taking into account the adjoint solution Φ* as well was the neutron source Q;

$$\Psi(r) = Q(r)\Phi^*(r)$$

The weighting function Ψ is normalized by the volume:

$$2\pi \int\int \Psi(r) r dr dz = 1$$

Then every homogenized material is determined by:

$$m_{\mathrm{hom}} = 2\pi \int_0^R \Psi(r) m(r) r dr$$

The one-dimensional geometry model comprising the homogenized weighed materials was applied for the cross-section generation of the fine-group library VITAMIN-B6 to the broad-group structure of BGL library. The new version BGLex was extended with cross-sections for the SS region.

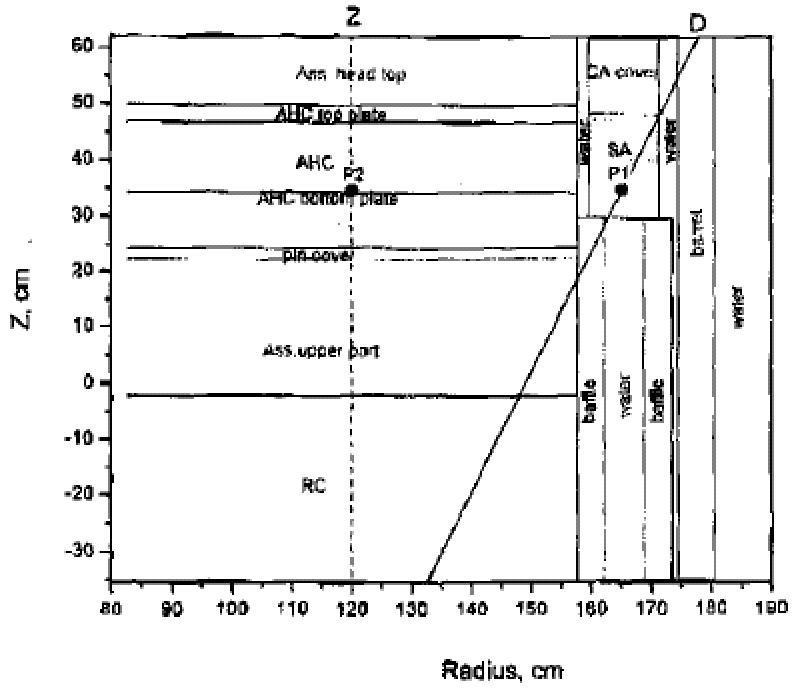

FIG. 1. Two dimensional R,Z - geometry model of the VVER-1000 reactor system

Verification and validation of the new version BGLex is forthcoming. It includes comparison between the calculated results with the new version BGLex and the libraries BGL and VITAMIN-B6 and comparison with experimental results.

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Containment Integrity – Key Role of Tendons

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In October 2009 in Praque, Czech Republic the CSNI Workshop on Ageing Management of Thick Walled Concrete Structures was held, the Nuclear Research Institute Rez was hosting the Workshop.

The safety significance of containment combined with current trend towards life extension and the regulatory authorities demands for even higher levels of safety assurance, means that ageing degradation must be effectivelly controlled, was observed as key conclusion of the Workshop. An important element of this control is an inspection and monitoring to assess and determine the condition of the concrete structures and associated components.

In our presentation we would like to concentrate on key element in the whole process, manufacturing and prestressing of tendons to be able to make meaningfull and validated force measurement and estimate time dependent prestressing force losses.

1. Improved manufacturing process of tendons
2. New improved prestressing machines and their calibration
3. Test tendons with new improved streching and fixation elements
4. Test bench for tendon tests
5. Options for prestress measurement to be tested
6. Test bench results
7. Future actions including research and development

Need of Reactor Dosimetry Preservation

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Today's nuclear renaissance in national and European aspect, expressed in building of new NPPs, as well as the development of Gen. IV nuclear reactors, meets new challenges of accuracy of the reactor analysis methods used for reliable operation and nuclear safety assessment. The nuclear safety requirements and philosophy have changed by the development of new nuclear systems and this imposes special research and development activity. Reactor Dosimetry (RD) which is applied for determination of neutron field parameters and neutron flux responses in different regions of the reactor system plays an important role in determining of consecutive effects from the irradiation. That is, for determination of radiation exposure on reactor system elements as reactor vessel, internals, shielding; dose determination for material damage study; determination of radiation field parameters for conditioning of irradiation; dose determination for medicine and industry application; induced activity determination for decommissioning purposes [1]. The management of nuclear knowledge has emerged as a growing challenge in recent years. The need to preserve and transfer nuclear knowledge is compounded by recent trends such as ageing of the nuclear workforce, declining student numbers in nuclear related fields, and the threat of losing accumulated nuclear knowledge.

Reinforcement of science and technology potential of many EU institutes is needed so to be able to support the nuclear operators and nuclear regulator in safety assessment as well as to strengthen the utilization of the research reactor for medicine and industry purposes.

The ways to preserve and develop the RD knowledge could be asked in the good practice of the near past within the European Working Group for Reactor Dosimetry (EWGRD), members of which are research organizations of the countries in Europe operating VVER, PWR and BWR type reactors. Joint workshops and training, commom intercomparisons will maintain the RD community experience and competency. Common research projects of the IAEA and EC will be a good base for development of common methodology as well as for involving more young researchers.

Young scientists and engineers urgently has to be attracted to the field of reactor dosimetry in order to transfer and further develop the available know-how. The interest of young researchers could be find between: receiving additional financial support, doing new professional contacts, involving in team work, involving in research/work community, creating feeling for usefulness and necessity, create feeling for proper pride.

The mentioned efforts for knowledge preservation will allow the RD to meet the demand of Gen IV reactors that is the RD to be used for determination of fast and epithermal neutron spectra, which will challenge materials performance with increased radiation damage. It will be applied as an important tool for growing number of reactors that will be decommissioned.

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Mobile Nuclear Laboratory for In-situ Measurements in NPPs

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In-situ gamma- and alpha-spectrometry, neutron counting and gamma-dosimetry are utilized in Paks NPP to assist maintenance operations as well as elimination of malfunctions with their consequences. Devices, methods applied and results, experiences achieved during the last 25 years will be described in the presentation. They are well applicable for the new generations of the future reactors as important additional safety measures. The Laboratory for Nuclear Safety and Techniques was established by the University of Debrecen and Quantechnologies Co in 2005 to utilize the results of the basic research achieved earlier in the investigations of fission, gamma-spectrometry and related fields.

Gamma-spectrometry of the primary loop pipes, ion exchange columns and steam generators is utilized regularly to all the 4 units of VVER-440/213 reactors at Paks NPP since 1985. (Three similar measurements were also carried out in Biblis-A, Germany.) The isotope selective, non-destructive analysis is applied to determine the activity of corrosion and erosion products as well as fission products. The measurements are carried out after the yearly shut-down with a cooling time ranging from several days to three weeks. Detected nuclides are: Cr-51, Mn-54, Co-58, Fe-59, Co-60, Zn-65, Zr-95, Nb-95, Ag-110m, Sb-122,124,125, Ru-103, Ru/Rh-106, I-131, Cs-134,136,137, Ba/La-140, Ce/Pr-144, Pm-148, Eu-154. Investigated locations include: 2-2 points at the hot-legs and cold-legs of each of the 6 loops, 8-10 heights on the ion exchange columns, 16-17 spots along the steam generator axis. Gamma-dose profiles are also determined. Shielded and collimated HpGe-detectors of different sensitive volumes (1, 3, 10, 100 cm3 and even a clover type of 4x100 cm3) are used for high-resolution, high-sensitivity gamma-spectro¬metry. Absolute full-energy peak efficiency for the diverse geometry (extended) sources are determined by experiments and calculations to produce final results as surface (or volume) activities. Electronics, data acquisition and evaluation are characterized by remote operation (cable length of 100 m), high counting rate (up to 200 kcps), PC-controlled digital signal processing. Results of the primary side measurements regard mainly the water chemistry and associated effects of the reactor operation by the corrosion/erosion nuclide activities and their long term behavior. (Fission product activities characterized the fuel-element hermeticity.)

An invaluable, detailed dataset was established during the 95 reactor-years the analyis of which is in progress. Some short-term and long-term observations were attained: differences and their time variations among the surface activities of the cold-legs and hot-legs reflecting some ageing behavior of the inner layers; effects of the auxiliary device operation like boron control circuit, deaeration of tap water supply, saturation of the ion exchangers; lost material; corrosion transport. Inspection of the decontamination process, its time evolution and isotope selective efficiency were also investigated. These results are essential in the normal operation and maintenance, forecast of the future contamination and dose levels, as well as for power upgrade and life-time prolongation programs.

A malfunction, caused during fuel casette cleaning, initiated the above techniques to be further developed and moved towards the in-situ uranium/transuranium isotope analysis. Radio¬active contamination of fuel assembly surfaces and pools (technical, cooling, fuel trans¬portation, etc) were analyzed by gamma- and alpha-spectrometry. The gamma-spectro¬metry was performed for the usual targets and for under water surfaces. In the last cases special containers („submarines”) were constructed with remote controlled variable size collimator-shutter. Spectrometers include some of the aforementioned HpGe detectors as well as CdZnTe-crystals of 2.25 cm3 sensitive volume. Although the energy resolution of the latter is worse than the formers (but better than that of scintillators), the high temperature operation and the appropriate geometry, small overall size make them extremely valuable for special measurements in hard environmental conditions.

Dry surfaces of the above mentioned pools and assemblies were analyzed by in-situ alpha-spectrometry, too. A cleanable Passivated Im¬planted Planar Silicon (PIPS) detector housed in a miniature chamber was fixed onto the surfaces by vacuum and some mechanical pressure. It measured alpha-spectra for the determination of surface activities from isotopes of Pu-238,239,240, Am-241, Cm-242,244. Experi¬mental correlations among rare-earth fission products (gamma-spectra) and transuranium nuclides (alpha-spectra) were established to make the latters’ activity to be estimated for non-attainable locations (eg. inner wall of steam generator heat transfer tubes). Similar cor¬relations with spontaneous fission neutron activity were also applied to observe transuranium (mainly Cm) isotopes from ”burried” sources.

These techniques and the appropriate mobile equipment (with the addition of PC-driven GM-tubes for beta-counting) can be applied to the inspection of high activity waste wells, waste deployment sites, nuclear industry, medical nuclear centers, environmental radioactivity (NORM, TENORM).

Risk Assessment for Nuclear Power Plants

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This paper is focused on the risk assessment for Kurskaya NPP.

The risk is deemed as a probability measure of hazardous natural or technogenic events. These events are accompanied by the occurrence and impact of adverse factors and resulting damage. The following types of damage are considered: physical, socioeconomic and environmental.

Incidents resulting from failures of technical equipment or its misuse are deemed to be the technogenic events (explosion, fire, etc.).

Climatic and geological hazards (earthquake, hurricane, tornado, etc.) are considered to be natural events.

The probabilistic risk assessment is based on the methodology of the probabilistic safety analysis (PSA) for nuclear power plants and other hazardous sites. We recommend the methodological approach of fault tree and event tree models. The fault tree / event tree models pertain to the large class of algebraic methods. This approach has a well-developed procedural framework, extensive databases and various verified codes.

According to Russian legislation, each hazardous site (a nuclear power plant, etc.) shall obtain a safety certificate.

We have made calculations for the safety certificate of Kurskaya NPP and assessed the risk factor system for the personnel of Kurskaya NPP, population and environment.

The output is the assessment of the risk factor system of Kurskaya NPP in physical and economic terms.

The physical values include the quantity of victims among the personnel and population, loss in agricultural products, etc. calculated taking into account the frequency (probability) of an incident. The economic values refer to the pecuniary losses corresponding to the physical values.

OPG/COG Equipment Reliability Initiative: Development of a consistent process to rate System and Plant Health at CANDU Plants

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System Health Monitoring has been in place at OPG Plants since early 1990's based on benchmarking from industry. A programmatic approach was taken. For each system (Important to Safety and Production), a comprehensive system performance monitoring plan was developed that defined the scope of the system surveillance that was performed. This included functional failure evaluation, trending of critical parameters and equipment/system margins. A system health report is issued semi-annually that reports on a number of indirect performance parameters e.g functional failures, backlog of work, status of PM program. The system is rated as follows:

GREEN: Excellent Performance
WHITE: Acceptable Performance
YELLOW: Needs improvement – Risk to Plant
RED: Unacceptable performance.

The rating was applied based on the judgment of the System Engineer and was used to drive System Health Improvement Action Plans especially for the RED and YELLOW rated systems. The rating of the systems however was based .

In 2005, Pickering B started the 85/5 Plan (improve capacity factor to 85% and FLR to 5%) and to get there a Material Condition Improvement Plan was developed. This plan was generated from the System Health reports and execution of this plan improved FLR from over 15% to about 5%. This was a good test of the integrity of the program.

In 2008 an initiative was proposed to standardize the monitoring of System Health across the COG Plants. This task was assigned to the author, the Program SPOC for System Health Monitoring at OPG.

In response to this, a team was setup that consisted of the System (Performance) Engineering Managers of each COG Plant including the three OPG sites. The team decided that to be consistent, a defined process was required that had less reliance on the judgment of the system Engineer. This process had to provide specific criteria for rating system health and had to be based on indicators that are standard to the COG industry.

This process would consist of the following:

1. Define Mandatory Leading and Lagging Indicators that will be monitored at all COG sites.
2. Define criteria for rating each indicator

3. Develop a method to perform an aggregate assessment of these indicators to determine system health
4. Define grouping of station systems that will be monitored and rated.
5. Develop a method to perform an aggregate assessment of all systems to provide a station score.

<u>Algorithm for Rating System Health:</u>

An algorithm based on an aggregate assessment of performance indicators was developed. The algorithm has leading and lagging mandatory indicators.

Leading Indicators include Outstanding Functional Failure Corrective Actions, Elective Maintenance Backlog –Online and Outage, PM Status, Operator Workarounds and Burdens, Modification backlog including TMODs and Temporary Configuration Changes. The weighting for leading indicators is 46% of the overall system rating.

Lagging indicators include Functional Failures, Corrective Maintenance Backlog –Online and Outage, Reportable Events, Open Technical Operability Evaluations and constitute 54% of the system score weighting.

Each indicator is assigned a weighting to reflect the importance of that indicator to system health and is rated at GREEN (3points), White (2 points), Yellow (1 point), RED (0 points). Score for each indicator is rating multiplied by weighting.

The system score is the sum of the scores for the leading and lagging indicators.

<u>System Health Rating thresholds are as follows:</u>

Green >80%, White (55-80%), Yellow (30-55%), Red (0-30%).

<u>Overrides:</u>

It was recognized that in some circumstances, under some circumstances a System Engineer should be able to over ride the computed system rating. However, to make this consistent, specific criteria were developed and the System Engineer was only allowed to override the system health downwards by one color rating e.g. White to Yellow.

<u>The criteria included:</u>

System being downgraded to RED for one reporting period if the system caused FLR > equivalent of one full day of production.

Aggregate assessment of site specified non-mandatory indicators. Eg On Line Fuelling Availability

<u>Aggregate Assessment of Station Systems - Plant Condition Index:</u>

Plant Condition Index (PCI) is an index used to provide an aggregate assessment of all monitored systems. The better the condition of the system, the higher the score. This index score is based on the color rating of all individual systems. Each system is weighted (based on importance) and a score assigned.

There are four groupings of systems – Special Safety Systems (Shutdown Systems) – 20% weighting, Emergency Core Cooling and Containment); S-98 systems – 30% weighting (systems important to safety), Fuel Handling Systems – 10% weighting, Systems important to safe/reliable production – 40% weighting.

It was recognized that PCI being a high level metric, it had to be consistent with Business Plan Objectives. It was determined that business plan objectives across COG utilities are to drive Special Safety Systems to GREEN (Excellent) and all other systems to WHITE (acceptable performance). Consequently, for Special Safety Systems, maximum points are awarded when system is GREEN. For all other systems, maximum points are awarded system is WHITE. Therefore a 100% score is equal to all Safety Systems GREEN and all other systems WHITE.

Implementation plans for each COG station are being developed that will result in a consistent approach to System Health Monitoring and a Station System Health score card. This methodology of rating system health can also be easily automated since most of the mandatory parameters are already monitored as part of station metrics. This would provide Station Management with an on-going assessment of system health so that attention can be focused on degrading systems.

On-line Gamma Spectrometric Monitoring in Water Cooled NPPs

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Development results and applications of automated HPGe spectrometers for precise on-line gamma spectrometric monitoring at the water cooled NPPs are presented.

Automated System for Radionuclide Monitoring of Coolant in Primary Circuit of NPPs based on convenient coaxial HPGe detector of 10% efficiency is intended for technological monitoring of the radionuclide specific activity in liquid and gaseous flows in the on-line mode.

The measuring unit is a U-shaped glass tube embracing the cryostate cover on the level of the detector center which is placed in a lead shield for protection against the background of external radiation. To ensure that there be constant flowing of the monitored medium, a bypass line is provided which connects the inlet pipeline with a system of organized through-flowing. A special algorithm is developed to carry out the measuring procedure consisting in cyclic washing of the measuring unit by water, taking measurements, and subtraction of the current value of background (residual) activity from the measured value of the controlled medium activity.

The results of measurements prove that the spectrometer is able to registrate efficiently the radionuclide specific activity in the total activity range in monitored liquid or gas flow up to 7.3×10^{6} Bq/l (2×10^{-5} Ci/l). The detection limit for the specific activity of the radionuclide ^{131}I is 1.8 $\times10^{3}$ Bq/l (5×10^{-8} Ci/l) at the measurement time of 600 s.

The automated spectrometer based on flowing HPGe detector with the through channel is intended for control of the uniformity of distribution of uranium and/or plutonium in fresh fuel elements, transferred through the detector, as well as for on-line control of the fluids and gases flows with low activity during the production cycle. The p-type HPGe crystal, which generally applied for the manufacture of the standard coaxial detectors with registration efficiency 10%, was used for the flowing detector manufacturing. The central through hole was made by the axis of the coaxial crystal with two open ends. The cryostat of the detector has the cover of a special design with the through channel of diameter 10 mm which comes via the through channel in the crystal. Thus, coming through the channel in the cover, made as the aluminum tube, radioactive sample (fuel element, fluids or gases flow) is found inside the germanium crystal and the registration geometry comes close to 4π-geometry.

The experimental curves of the registration efficiency of the developed flowing and standard coaxial detector of the similar volume are presented. The flowing detector has efficiency registration of gamma quanta with energy 200 keV 10 times higher, with energy 80 keV 20 times higher but with energy 40 keV – 70 times higher. At the same time the lower limit of the energy range for the developed flowing detector was 20 keV compare to 40 keV for the standard coaxial detectors based on HPGe p-type crystals.

Automated System for Water Activity Measurement in Outlet Channels of NPPs comprises two or more floating monitoring stations, placed in outlet channels and information reception station, situated in administration building of the NPP. The system is self-diagnosable and provides on-line quantitative analysis of gamma-spectra for more than 30 nuclides of low activity level and subtraction of radionuclide background containing in monitoring medium. Detection limit for volumetric activity for radionuclide Cs-137 at measurement time 1 hour does not exceed 115 Bq/m^3. The limit of permissible relative error for Cs-137, measurement time 1 hour and value of volumetric activity not less than 2000 Bq/m^3, does not exceed ± 16%. The instability of volumetric activity measurement from calibration source does not exceed ± 10%. The upper limit of the measuring activity is not less than 4×10^8 Bq/m^3.

Point Lepreau Level 2 PSA Results, Insights & Risk Monitoring Program

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A Probabilistic Safety Assessment (PSA) for Point Lepreau Generating Station has been completed as part of the plant Refurbishment Project. The main objective of this PSA is to provide insights into plant safety design and performance, including the identification of dominant risk contributors and assessing options for reducing risk. The scope of this assessment covers Level 1 and 2 PSA and includes internal events for full power and shutdown, internal fires and internal floods as well as PSA-based seismic events margin for full power operation,

The Point Lepreau Refurbishment (PLR) Probabilistic Safety Assessment (PSA) is the first Level 2 PSA for both internal and external (fire, flood, and seismic) events performed in Canada. Similar recent CANDU® 6 PSAs include a Level 2 PSA for internal and external events for Wolsong 2, 3, and 4 and a Level 1 PSA for internal events for Qinshari CANDU® 1 and 2. Following the lead of Point Lepreau. Gentilly-2 has also begun the work for a Level 2 PSA for internal and external events.

Both Atomic Energy of Canada Limited (AECL) and New Brunswick (NB) Power PSA analysts participated in the PSA. For most systems, NB Power provided reliability models and AECL incorporated common cause failures (CCFs) using the Unified Partial Method (URM) [1] and human reliability analyses (HRA) using Accident Sequence Evaluation Program (ASEP) [2]. All PSA methodologies and analysis reports were reviewed by NB Power. Component failure and operation data was provided by NB Power for input to the analysis models.

The Level 2 PSA was performed in order to evaluate the summed severe core damage frequency (SCDF) and the summed large release frequency (LRF) for the refurbished Point Lepreau plant. For Level 1 Accident Sequence Quantification (ASQ), mitigating system fault trees were merged into a single master fault tree in order to quantify the event trees. Mutually exclusive events were removed and recovery factors were applied to provide a realistic estimate of the SCDF. The dominant Level 1 severe core damage sequences were grouped based on similar core damage progression. For each grouping, severe accident progression analysis was performed. Using these results, the containment event trees (CETs) were developed. Containment FTs were merged into the Level 1 master fault tree to quantify the CETs and estimate the LRF. The Level 1 and Level 2 sequences are fully linked for all events (internal events, internal fire, flood and seismic).

Following ASQ for internal events, fire and flood, the results were integrated to provide an overall estimation of the SCDF and LRF for the refurbished Point Lepreau plant. In the process, a number of potential changes (design, maintenance, testing) hase been identified. Some of them are essential in order to meet the risk limits for the severe core damage and external release frequencies. The combined result meets the risk limit of 1E-04/yr and 1 E-05/yr for SCDF and LRF respectively (Reference [3]). which is in line with the international results for the refurbished plants. Also the results for the PSA-based Seismic Margin Assessment meet the target of 0.3g.

Using the integrated results, importance analysis was performed to identify risk-significant failures, with Fussell-Vesely and Risk Achievement Worth indices, and risk-contributors with Risk Reduction Worth indices. Based on the importance measures, analysis was performed to evaluate the sensitivity of the SCDF and LRF results to the dominant contributors. Uncertainty analysis was also performed to provide qualitative discussions and quantitative measures of the uncertainties in the results of the PSA, namely the frequency of severe core damage or external releases. Based on the results, recommendations were made to improve maintenance, testing, training procedures as well as housekeeping. Currently, many of these recommendations are being implemented, while others are being evaluated with additional benefit-cost analysis.

With the completion of the Level 2 PSA, an Operational Risk-Informed Management Program (ORIMP) has commenced to develop an online risk monitoring tool which will provide risk-insights and enable risk-informed decision making at the plant. As a part of ORIMP, knowledge gathering and benchmarking of industry practices are underway. Also, modifications are necessary to tailor the PSA models for an operational application. Initially, the online risk monitor tool will encompass Level 1 internal events only and as such, measure the severe core damage frequency. Subsequently, the scope will be expanded to include Level 2 and measure the external release frequency.

This paper discusses the results and insights gained from the Level 2 PSA and the transition to the risk monitoring program.

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Station Blackout Core Damage Frequency Reduction at LABGENE Reactor -The Contribution of a Natural Convection Residual Heat Removal System

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This paper presents an evaluation of the Station Blackout-Core Damage Frequency (SBO-CDF) for the LABGENE reactor. The contribution of a passive residual heat removal system in reducing the SBO-CDF is investigated. Systems and devices involved in SBO are described and their influence on accident sequences are discussed.

LABGENE is a 48MWth PWR prototype reactor being designed in Brazil. This reactor is intended to serve as a test bed for developing the capability to design small and medium size power reactors for electricity production and for nuclear propulsion. The total cost of LABGENE is estimated to be around US$ 488 million. So far US$ 318 million have been spent in this project. Some of the main components of the reactor have been delivered to the site, namely the pressure vessel and internals, the pressurizer, the two steam generators, primary pumps, and components of the secondary circuit [1]. The reactor site is located in a rural area 120 km from the city of São Paulo, Brazil.

The reactor has two primary coolant loops. The pressure vessel, steam generators, the primary pumps, and the pressurizer are enclosed in a steel containment, which is surrounded by a water pool used as shielding and heat sink. A confinement building houses the steel containment and a secondary system with two turbo-generators.

LABGENE has two independent systems to remove decay heat from the core: A forced circulation (FCR), and a natural circulation system (NCR) that exchange decay heat with the shielding pool. The NCR system has four bistable valves driven by a DC Electric System. The DC power is supplied by emergency batteries .

Station blackout at LABGENE is defined by the following sequence of events: loss of electric power provided by the turbo generators, concurrent with the unavailability of emergency on-site power(EDGs) , and the loss of the off-site power sources.

It was found that the initiating event SBO has an estimated frequency of 7.1E-05/year. A major contributor to this frequency is the loss of turbo generators. The loss of the electric supply from the external grid is the second major contributor.

The SBO-CDF uncertainty associated with the basic data was quantified. The results are shown in FIG. 1 for the two design configurations analized. As shown the natural circulation system system has a significant impact on the SBO-CDF.

A sensitivity analysis was performed to identify those systems, actions or components that most influence the SBO-CDF. It was shown that decreasing the failure probability of Electric Diesel Generators(EDG) by a factor of ten, the corresponding SBO-CDF decreases from 6,0E-09 to 6.6E-10. If the performance of the EDG is degraded by a factor of ten, then the SBO-CDF increases from 6,0E-09 to 1.8E-06. This behavior can be explained by analysing cut sets of the SBO fault tree. In fact, an SBO requires different combinations of commom cause failures of the four EDGs. Depending upon the number of combinations, the SBO-CDF increases by powers of two, tree or four. Reducing the EDG reliability does not reduce the SBO-CDF in the same way, in this case other failures mode become significant contributors.

An analysis was also performed to quantify changes in the SBO-CDF due to changes in EDG outages for test and maintenaince. The results show that increasing the number of outages, so that the reliability associated is reduced by a factor of ten, the SBO-CDF increases proportionally. On the other hand, decreasing the number of outages in a way that the reliability associated is increased by a factor of ten, the SBO-CDF decreases only four times.

Situations in which the operator fails to align the AAC system can increase the SBO-CDF by a factor of five. Improving the ability of the operator by a factor of ten the SBO-CDF can be reduced by a factor of four.

The reliability of the NCR system is heavily dependent on the bistable valves.The contribution of these valves to the failure of the system is about 93 % .

The SBO-CDF was compared with data presented in NUREG/CR-6890 [2]. The design configuration with the NCR is well placed having an SBO-CDF mean equal to 6.1E-9/year. Without the NCR the mean is 2.9E-6/year(FIG. 2).

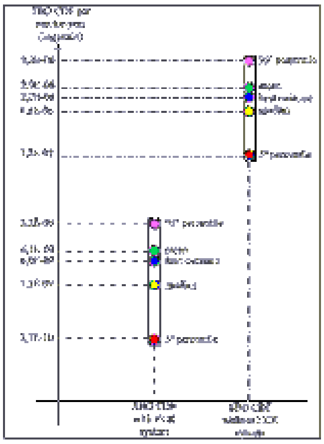

FIG. 1. SBO-CDF Uncertainties

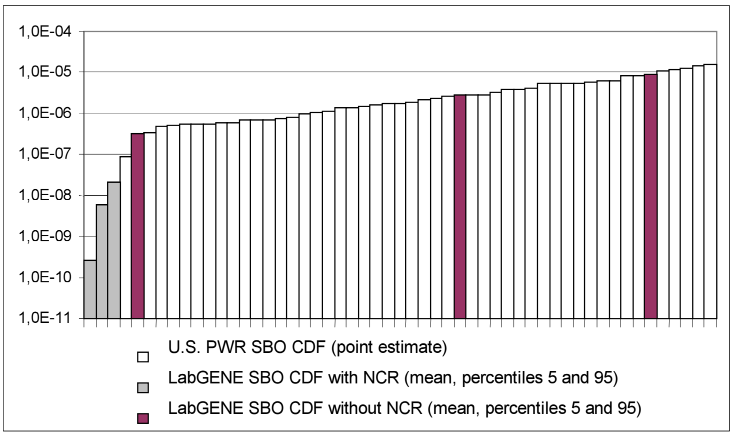

FIG. 2. SBO-CDF LABGENE vs NUREG/CR-6890

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R&D of a Chinese SCWR

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This paper briefly describes the long-term R&D program for a Chinese SCWR issued by NPIC and introduces the related R&D work performed by NPIC. A technical description is also provided on the Chinese SCWR conceptual design and demonstration.

The Chinese SCWRR&D can be divided into three phases and consists seven interrelated work tasks.

Phase 1：Technical demonstration and basic technology R&D, including：

Task 1：Top-tier requirements and general planning

Task 2：Initial feasibility study

Task 3：Basic technology R&D

Phase 2：Conceptual design and engineering technology R&D, including：

Task 4：Conceptual design and feasibility study

Task 5：engineering technology R&D

Task 6：Basic design

Phase 3：Detailed design of the prototype reactor, including：

Task 7：Detailed design

The main technical parameters of the Chinese are shown in the following table.

Items	**Values**
CDF	Lower than Gen-3 NPP
LIRF	Lower than Gen-3 NPP
Core thermal margin	≥15%
Output electrical power	1000MWe ~ 1700MWe
Plant thermal efficiency	~ 44%
Fuel cycke	18-24 months
Design life	60 years
Fuel enrichment	＜8%
Capital cost	Lower than Gen-3 NPP

Now the demonstration work is being performed for the core/fuel assembly and engineered safety systems.

Verification of Advanced Design Features Adopted in an Integral PWR

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Small and Medium sized Reactors are currently under development worldwide not only for electricity generation, but also for sea water desalination. SMART [1-2] is an integral PWR with a sensible mixture of new innovative design features and proven technologies aimed at achieving highly enhanced safety and improved economics. In the beginning stage of the SMART development, top-level requirements for safety and economics were imposed for the SMART design features. To meet the requirements, highly advanced design features enhancing the safety, reliability, performance, and operability are introduced in the SMART design. The enhancement of safety is realized by incorporating inherent safety 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.

SMART design combines firmly established commercial reactor design technologies with the new advanced technologies. The advanced design features require tests to confirm the performance of design/engineering and major components. Although most of the technologies and design features implemented into the SMART have already been proven in the relevant industries, the new advanced design features should be proven or qualified for use in the SMART design. The SMART design verification program includes development of the analysis models, basic thermal-hydraulic experiments, comprehensive experiments, and tests of the major components.

In coupling a desalination plant to a nuclear energy supply system, the economic consideration is emphasized for the selection of the desalination process requiring the least energy for the target water production and thus the generation of more electricity under given conditions. For the economic choice of the desalination process with the target water production, MED-TVC and a steam extraction from the turbine were selected for the coupling with SMART. The most important safety concern in using nuclear thermal energy for desalination is the radioactivity carry-over into the product water from the nuclear reactor. In the integrated nuclear desalination plant, several events were identified as the potential disturbances of SMART imposed by the desalination plant such as a main steam flow increase and a decrease in the feed water temperature. The impact of these disturbances on the Design Basis Accidents and Performance Related Basis Events of SMART were evaluated. The safety of the SMART design is assessed for more than 1500 design based events of 21 different types. The computer code used for the analysis is MARS/SMR, which is a best-estimate thermal-hydraulic system analysis code based on a two-fluid model for two-phase flows. The safety analyses were carried out for the SMART basic design by using conservative initial, boundary conditions and assumptions. The results also show that the key safety parameters of any potential disturbance of the integrated nuclear desalination plant do not violate the specified limits of SMART.

Various thermal hydraulic tests were conducted to support key technology and design verification of the SMART systems. Fundamental thermal-hydraulic experiments were carried out during the concept development to assure the key technology of the advanced safety systems. During the SMART design/engineering verification phase, various separate effect tests and comprehensive integral tests were conducted. The separate effect tests examine the behavior of particular components. The data from the separate effects tests are being used to develop and verify the safety analysis models. While the data obtained from the integrated system effect tests [3] are being used to verify the capability of the analysis method and predict the system characteristics of the integrated innovative safety systems. Thermal hydraulic behavior for operational transients and design basis accidents has been experimentally investigated using the thermal-hydraulic integral test facility, VISTA (Experimental Verification by Integral Simulation of Transients and Accidents). The thermal-hydraulic responses following the design basis accidents are experimentally examined and they are used to verify the system design of the SMART.

This paper describes verification program of advanced design features adopted for the SMART systems including computer analyses, thermal hydraulic tests and performance tests of major components.

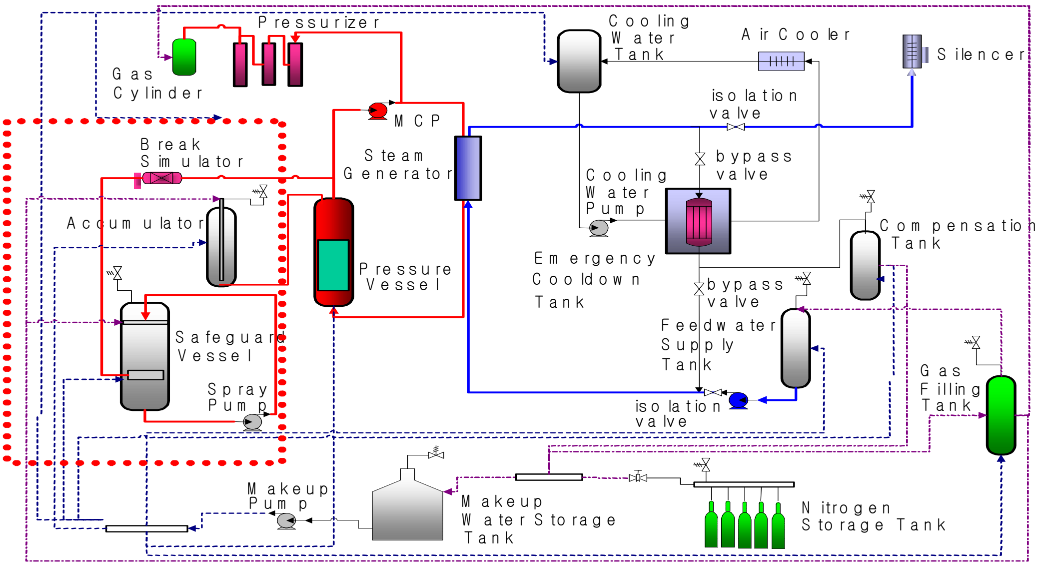

FIG. 1. Schematic diagram of the VISTA

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Several Aspects on Material Problems for SCWR

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The supercritical-water-cooled reactor（SCWR）system is being evaluated as a Generation IV concept because it builds on currently proven light water technology to provide for high thermal efficiency and plant simplification. Materials selection is a critical gap that development and a long term safety operation of the supercritical-water-cooled reactor（SCWR）need to be faced with. Design concept of increased temperature and pressure, radiation and supercritical water coolant brings into a more aggressive environment to candidate materials. ASME rules provide fundamental needs with various aspects as a guideline on nuclear power plant (NPP) design. This paper covers materials design requirements in ASME code section III subsection NH for elevated temperature components in NPP from the point of view at engineering design angle. Establishment of new codes and standards is needed for the deployment of materials in SCWR in use. Multidisciplinary cross research and application of advanced computerical material science in screening suitable materials and building in predicable performance in plant operating experience are recounted and proposed. In addition, the present situation and existing problems on industry production of candidate materials in China are introduced.

Generation IV reactor, IRIS

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Nuclear Power Reactors can be Classified to four generations: generation I reactors, generation II reactors, generation III reactors and generation IV reactors.

Generation II reactor is a nuclear reactor of one of several types developed from the first generation I reactors. The change was great but not entirely revolutionary, with some completely new types and technologies being used. Prototypical generation II reactors include the PWR, CANDU, BWR and GCR [1, 2].

A generation III reactor is a development of any of the generation II nuclear reactor designs incorporating evolutionary improvements in design which have been developed during the lifetime of the generation II reactor designs, such as improved fuel technology, passive safety systems and standardized design.

Generation III reactors such as: The Advanced Boiling Water Reactor (ABWR), a GE design which first went online in Japan in 1996, the AP600, a Westinghouse Electric Company design which received final design approval from the NRC in 1998 and the European Pressurized Reactor (EPR).

Generation III+ designs are generally extensions of the Generation III concept which include advanced passive safety features. These designs can maintain the safe state without the use of any active control components. These includes: The Advanced CANDU Reactor (ACR), the AP1000, based on the AP600, the Economic Simplified Boiling Water Reactor (ESBWR), based on the ABWR, and the APR-1400, an advanced PWR design evolved from the U.S. System 80+ which is the basis for the Korean Next Generation Reactor or KNGR [3, 4].

Generation IV reactors (Gen IV) are a set of theoretical nuclear reactor designs currently being researched. Most of these designs are generally not expected to be available for commercial construction before 2030, with the exception of a version of the Very High Temperature Reactor (VHTR) called the Next Generation Nuclear Plant (NGNP). The NGNP is to be completed by 2021.

Current reactors in operation around the world are generally considered second or third generation systems. The first generation systems retired some time ago. Research into these reactor types was officially started by the Generation IV International Forum (GIF) based on eight technology goals. The primary goals are: improve nuclear safety, improve proliferation resistance, minimize waste and natural resource utilization, and to decrease the cost to build and run such plants. The Evolution of Nuclear Power Plants with time can be seen in FIG. 1.

Today there are 441 nuclear power reactors in operation in 31 countries around the world. Generating electricity for nearly 1 billion people, they account for approximately 17 percent of worldwide electricity generation and provide half or more of the electricity in a number of industrialized countries. Another 32 are presently under construction overseas. Nuclear power has an excellent operating record and generates electricity in a reliable, environmentally safe, and affordable manner without emitting noxious gases into the atmosphere.

There are many innovative (Generation IV) reactor designs in the words. One of these reactors is the International Reactor Innovative and Secure (IRIS). The IRIS is a Generation IV reactor design made by an international team of companies, laboratories, and universities and coordinated by Westinghouse. IRIS is hoped to open up new markets for nuclear power and make a bridge from Generation III reactor to Generation IV reactor technology. The design is not yet specific to reactor power output. Notably, a 335 MW output has been proposed, but it could be tweaked to be as low as a 100 MW unit. IRIS is a smaller-scale design for a Pressurized water reactor (PWR) with an integral reactor coolant system layout, meaning the steam generators, pressurizer, control rod drive mechanisms, and reactor coolant pumps are all located within the reactor pressure vessel. This causes it to have a larger pressure vessel than an ordinary PWR despite a lower power rating, the size is more comparable to that of an ABWR as can be seen in FIG. 2 and FIG. 3 [1,2].

Advantages

Most of the advantages of the new IRIS design are safety related, although Westinghouse claims that IRIS will be able to deliver power at competitive rates as well. Due to Economies of scale, modern nuclear plants tend to be built with larger electrical outputs, such as the European Pressurized Reactor, which has scaled up power to 1600 MW in new plants. IRIS, on the other hand, is built to be used in countries where there are not extremely large electric power grids, mainly developing nations. Due to limitations on power of individual power stations versus total grid size, plants who's power is over a certain percentage of grid size are infeasible in such situations.

Due to simplifications and greater safety, it is believed by Westinghouse that in spite of its size, analysis estimated a target total cost of electricity at about 4 ¢/kWh. Given its small power and physical size, it is expected that multi-unit sites could be operated efficiently, Westinghouse estimates that a 3-unit site could be built in 9 years with a maximum cash outflow of 300 M$. One cost saver, for instance, is the need for only one control room, from which all units at a multi-unit site can be controlled [1,2].

Aside from economics, these are a few other advantages that the IRIS has:

— **Fewer penetrations to the pressure vessel -** by having the control rods and all drive mechanisms contained within the vessel, the need for dozens of small penetrations is eliminated, which are extremely costly. The only penetrations used are for the incoming and outgoing secondary coolant and for emergency safety systems.

— **Large operating margins -** the operating margins are typically the measure of a value compared to what that value would have to be to fail the fuel. IRIS effectively gets much lower operating margins by having a core with a much lower power density, while the core is mostly the same size as a current PWR, the thermal output is much smaller, making it much less likely to reach film boiling and fail in an accident.

— **Lower radiation doses to workers -** due to the confinement of all the RCS components and more shielding (by a larger water mass) result in low estimated doses for plant workers than current designs.

— **Collaboration and research -** incorporating so many universities and labs into the project is expected to have a number of benefits, one is contributing to the academic knowledge available for new plants, another is that researchers in many diverse countries with experience regarding the IRIS will be useful when they are deployed, because a goal of the project is to eventually build plants in countries that do not currently have nuclear plants.

— **Lowered core damage frequency (CDF) -** as a result of all the individual innovations that improve safety and an in depth Probabilistic risk assessment study that refirefine the net safety risk, IRIS has the lowest CDF (which is a quantitative measure of the probability of a major core accident taking place) associated with any proposed plant of 10^{-8}.

— **Marketing and licensing -** With the vastly improved safety, there should be a quick and easy licensing associated with the design, and it could occupy a large part of a growing market for small size nuclear power plants.

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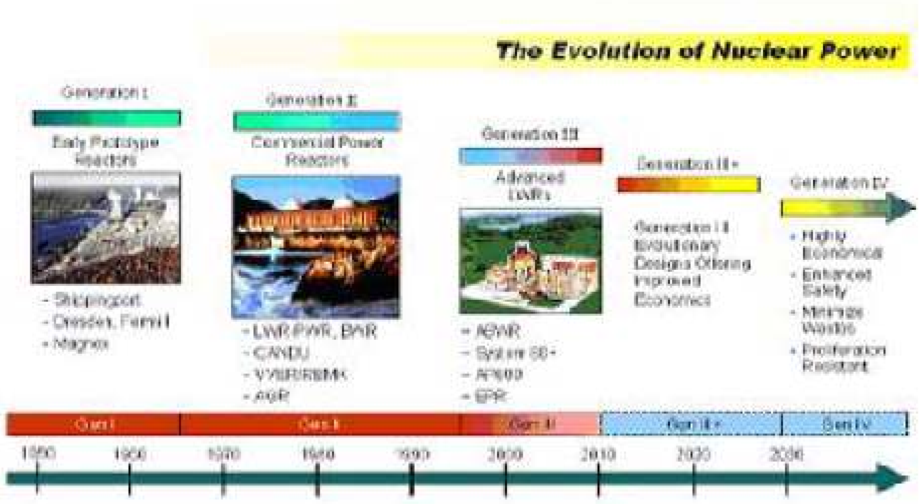

FIG. 1. The Evolution of Nuclear Power Plants with time

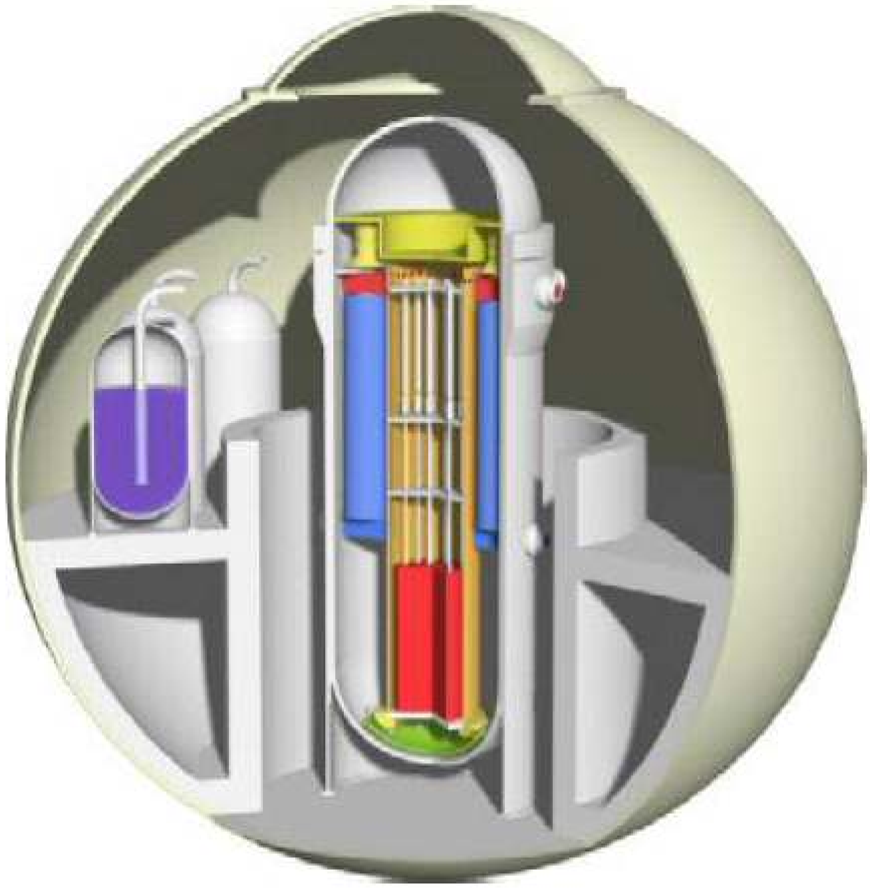

FIG. 2. Components of the 335 MWe (IRIS) reactor

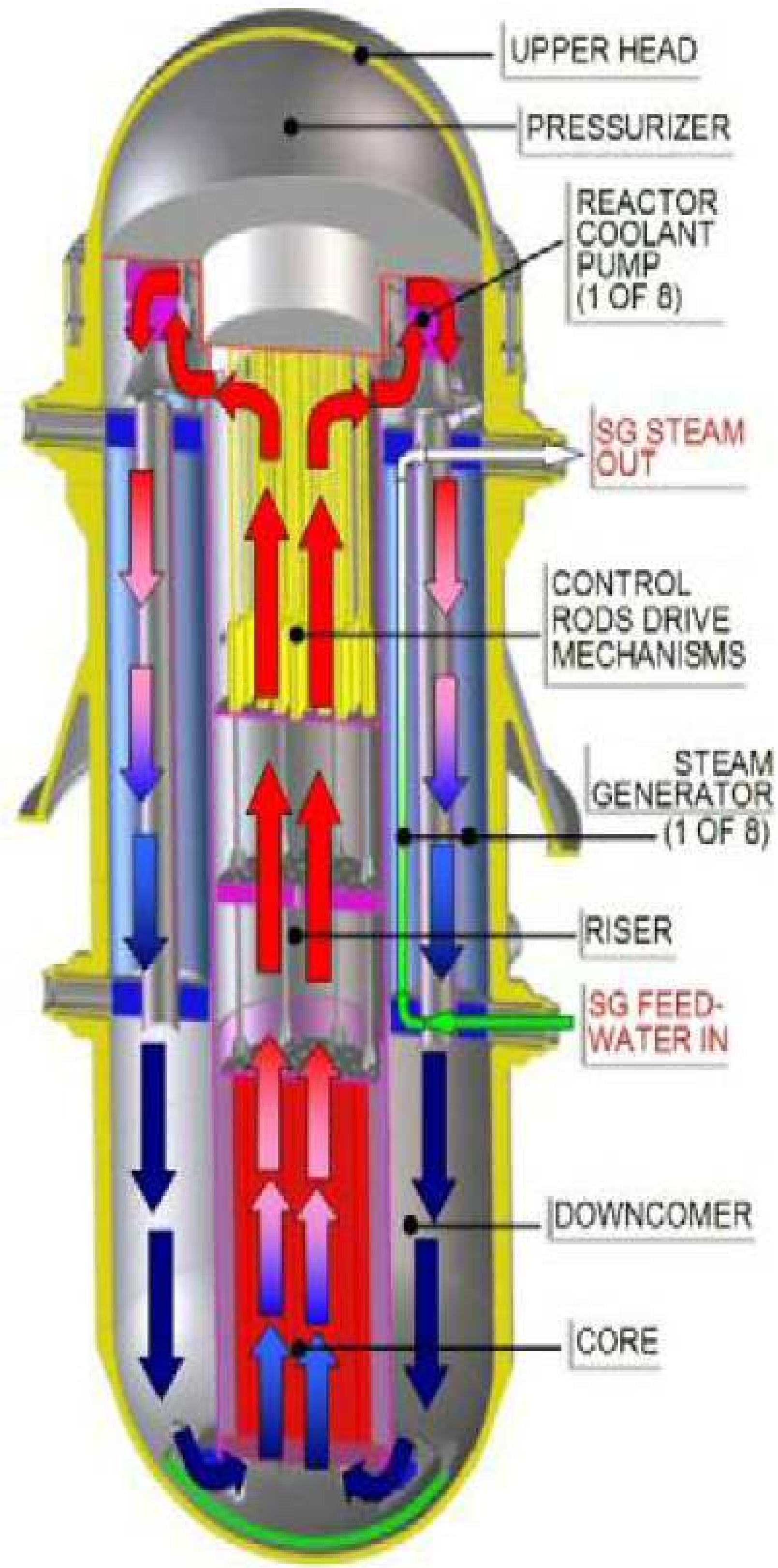

FIG. 3. IRIS Integral Primary System

Model-based Advanced Water Cooled Reactor

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At present, no easy-to-maintain, high-temperature capable monitoring and control systems exist for Generation-IV reactors. Since these reactors are planned to operate with temperature above 1000^0C and high neutron flux, accurate measurements are needed to identify hot spots, control in-core power distribution and other parameters that are critical to the safe operation of the reactors. Also, an on-line gamma-spectroscopy approach is the most accurate method to perform the fuel monitoring, especially for reactors with on-line refueling. It is important to operate a reactor in the minimum fuel loading state so that any physical changes in configuration are self-correcting.

A process to safely convert the most hazardous waste products of industrial society (carbon dioxide, plutonium and depleted uranium) into emission-free fuel, pioneered by the WETC, will significantly reduce global warming and the threat of nuclear terrorism. The process, involving a high flux sub-critical reactor (HFSR) and a proliferation-resistant fuel cycle, will avoid keeping the gaseous fission products restrained in the fuel rods, which is an innovative strategy used to monitor and control the reactor. They could then be stored as solid products in well-protected containers outside the reactor core.

This design will develop an in-core power monitoring system, which will rely on two forms of the same application: hardware means and a physical model that calculates the reactor parameters by dividing the core into a plurality of cells. Since the neutron flux in any cell is independent of whether it is produced by a large number of neutron sources simultaneously (as in the reactor) or by a single source placed in each cell, an inexpensive test facility could be used to examine hardware and software.

In this approach, the fuel element simulators have the same configuration and dimensions as the reactor’s fuel assemblies. The fuel channels are filled with a neutron absorber and depleted fuel, which can be in the form of particles, annular pellets or liquid. A miniaturized fission chamber and a loop filled by gas or steam simulate delayed-neutron and gamma ray emitter diffusion and feedback system performance.

Several detector packs, each includes ^{232}Th, ^{235}U and uncoated devices are distributed through a plane in the core. A small source of neutrons moves along the various channels to simulate the core nuclear characteristics, including the reactor’s critical mass and neutron flux distribution. Prototypes of the control system that included some aspects of this design were tested in the mid 1960s and 1980s.

The HFSR is comprised of coaxial neutron and energy-amplifying regions separated by moderating and thermal neutron absorbing layers. Emphasis is placed on thermal region employing depleted or natural uranium as the fuel material. The sub-critical fast and thermal regions are coupled together to form a neutron amplifier. According to Avery, the coupled two-zone reactor kinetics is determined by four integral parameters that are denoted by k_{11}, k_{22}, k_{21} and k_{12}. Here k_{11}, k_{22} are the multiplication factors of the zones 1 and 2 on their own, k_{21} and k_{12} are coupling coefficients.

The amount of fuel is such that each region is always sub-critical. A blanket multiplication factor of k_{22} is not greater than about 0.95 and a booster multiplication factor of k_{11} is not greater than about 0.98. The main effect of the delayed-neutron emitter circulation on reactivity is in the increasing role of delayed neutrons. In the traditional approach, whereas their coupling appears through the interface boundary conditions, we need only one corresponding parameter, k_{eff}. If we consider k_{21} as a parameter, it can be shown for the stationary state that the overall gain of the blanket is approximately equal to $A/(1-Ak_{21})$, where $A= k_{12}/(1-k_{11})(1-k_{22})$.

Also, a new concept of high efficient neutron generation to control the reactor axial power distribution is examined. A class of dielectric structures has been proposed that would provide a high efficiency when excited by a charged particle beam. The analysis takes into consideration a wide range of the neutron generator design aspects including one of the early assumptions of the theory of charged particle acceleration that, in electrodynamics, the vector potential is proportional to the scalar potential and high density charge clusters confined by self-consistent electric field.

The preliminary analysis shows that modification of current water-cooled reactors is cheaper and safer than storing spent fuel. In addition, our test facilities will either use the General Atomics' inherently safe fuel technology or will not employ any fissionable material. Relationships will be developed between the key design parameters including system size, power level, and neutron flux. The advanced annular fuel design could be initiated in collaboration with Massachusetts Institute of Technology, as was previously discussed among several MIT researchers.

CANDU: Setting the Standard for Proliferation Resistance of Generation III and III+ Reactors

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The nuclear renaissance will bring new safeguards challenges before the IAEA in the form of non-traditional fuel cycles and system designs, in states that may or may not have an existing nuclear program. Thc considerable near-term build campaign of Generation III and III+ reactors will necessitate a reliance on proven technologies and processes wherever possible.

The CANDU reactor is currently the most comprehensively safeguarded power reactor in operation, due in part to its status as an on-load refuelled power reactor. Of equal importance are the inherent features of CANDU reactors that contribute to its overall proliferation resistance, including those that enable an efficient and effective safeguards approach - a characteristic referred to as "safeguardability". The safeguardability of the CANDU reactor (represented in the Gen III and Gen III+ classification as Enhanced CANDU 6 and ACR-1000, respectively) can be traced to a number of inherent proliferation barriers and safeguards-enabling features that derive from the fundamental physics of heavy-water power reactors.

In its irradiated fuel, a CANDU reactor (whether Enhanced CANDU 6 or ACR-l000) produces roughly half the plutonium per unit weight than that of an I.WR, and with a comparably low (reactor-grade) purity of fissile plutonium. Moreover, to acquire a weapons-usable quantity of fissile material one would need to divert between 100 and 200 highly radioactive CANDU fuel bundles, weighing 2-4 tonnes without shielding and requiring 100-200 successful defeats of safeguards, followed by remote reprocessing of a highly radioactive material that is relatively dilute in plutonium.

The on-load refuelling of a CANDU reactor is necessarily a highly controlled and automated process (one cannot use human operators directly in this process due to the high radiation fields at the face of the reactor). The inherent transparency of this operation makes it relatively easy to track and verify fuel movement through the refuelling process, and also detect off-normal movements of the fuelling machine such as unusually frequent visits to selected fuel channels. Combined with the historically low defect rate of CANDU fuel (<0.1%), this presents an inherent barrier to the attempted diversion of fuel by disguising it as defective fuel.

On-load refuelling also leads to spent-fuel bumups that fall within a relatively well-defined band, as each fuel bundle experiences irradiation at various axial positions during its residence in the core. This leads to a high probability of detecting off-normal burnup values.

In addition to being easily detected through monitoring of fuel movement, the potential misuse of a CANDU reactor required for the type of rapid refuelling necessary to create weapons-grade (or near-weapons-grade) plutonium is thwarted by both engineering and physics constraints that are fundamental to heavy-water power reactors. The fuelling machines are highly complex, automated machines designed to operate reliably at a designed duty cycle, and are not capable of sustained operation at significantly increased rates. This affects not only the capability for rapid refuelling of selected channels, but also the capability to maintain regional overpower margins, and ultimately core reactivity. Thus the inherently low excess reactivity of the CANDU core, and the highly complex and mission-oriented nature of the fuelling machines designed to maintain steady core power under these conditions, combine to discourage misuse of the core significantly beyond the design envelope. This is a significant difference between a CANDU power reactor and on-load refuelled reactors designed for military plutonium production.

Finally, CANDU fuel cycles, employing natural uranium for Enhanced CANDU 6 and 2.4% enriched uranium for ACR-l000, do not require enriched uranium in quantities that neccesitate the establishment of an indigenous enrichment capability. In the case of ACR-1000 this remains largely a state-level decision however. At the same time the fuel cycles for the Enhanced CANDU 6 and ACR-1000 reactors leave spent fuel that is, as mentioned, relatively sparse in fissile material. Thus, there exists relatively less incentive for a State to acquire a reprocessing capability solely related to its operation of CANDU power reactors, although as with enrichment this remains a state-level prerogative. In a state with predominantly CANDU technology, the significance of either of these signature activities is heightened.

When the total fuel cycle is taken into account, the absence of a need for enrichmcnt technology for Enhanced CANDU 6 lowers not only the proliferation risk of this system, but also the cost of applying safeguards, particularly to the front end of the fuel cycle.

Historically, civilian nuclear power reactors under international safeguards have not proven to be attractive targets for nuclear weapons proliferation, and CANDU technology has a proliferation resistance that is second to none. Combined with its safety record and fuel-cycle flexibility, CANDU technology provides an attractive platform for meeting the evolving global need for safe, secure, and non-polluting energy supply.

IRIS: Advanced Water Cooled Reactor Technology

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IRIS has been in development since the turn of the century and the subject of hundreds of presentations and publications. Thus, the purpose of this paper is not to detail IRIS characteristics, but rather to point out its advanced approach to the themes of this conference: design, safety, operation and maintenance, construction, and innovation applications. It will also point out how IRIS is very attractive for countries approaching nuclear power for the first time in terms of technology, economic, human resources and infrastructure.

IRIS is a 335 MWe PWR of the integral design, where the entire primary system is within the reactor vessel. Gone are the external to the vessel steam generators, pumps, pressurizer, together with the associated large piping and the large forged vessels. IRIS has only one forged component (the primary reactor vessel) versus the dozen or so for other LWRs. Gone together with the large piping is also the possibility of large LOCAs (Loss of Coolant Accidents) and thus the need for safety systems designed to cope with an accident which cannot occur anymore. These are just examples; in fact the philosophy of the IRIS design, made possible by the adoption of the integral configuration, is to emphasize simplicity, which begets increased safety and decreased cost.

The safety approach is centered on the safety-by-design™ which means that accidents are eliminated by design, rather than coping with their consequences. For those accidents which cannot be eliminated, then their consequences and probability are minimized, again by design. The bottom line is that IRIS has: only five simple and relatively inexpensive passive and zero active safety systems; only one accident (refueling) capable of producing a Class IV accident (versus eight for "traditional" LWRs); and, a core damage frequency for internal events equal to 2E-8. This allows IRIS to reduce the Emergency Planning Zone (EPZ) to 1 Km or less, essentially the plant boundary, which in turns allows siting of IRIS near population centers. This is very important for cogeneration applications like desalination, district heating or even bio-fuel generation, which require the power plants to be near the user. The IRIS control system has been specifically designed to allow variable cogeneration.

The simplicity approach of IRIS is reflected not only in design, but also in operation and maintenance. Routine maintenance is needed only at 48 months' interval and IRIS can accommodate several core designs with refueling intervals from about 36 months (which give the maximum burnup allowed) to 48 months (which gives the highest capacity factor). In IRIS the vessel fluence is five decades less than in other LWRs, below the NRC threshold for surveillance, yielding less operational cost, less personal radiation, no vessel lifetime limitation as far as radiation is concerned and, simplified decommissioning.

Another IRIS innovative design feature is its compact design: there is only one single NSSS building, which includes the auxiliary, containment and control buildings in one structure of a little more than 50 m diameter, with positive effects on cost and security. This simple, compact building is very economically installed on seismic isolators, thus basically eliminating seismic from the accidents spectrum.

The cost of IRIS is extremely competitive with the large LWRs. First of all, the different and simplified IRIS design does not belong to the same economy of scale curve of traditional LWRs, but to one roughly parallel and lower (in our current estimate about 20-25% lower). Additionally, IRIS has the economy of multiples, realizing savings by multiple units on the same or different sites, learning, mini-serial components production (for example a 1700 MWe 3-loop PWR has three massive steam generator and pumps, while series of IRIS multiples with the same total power have forty small and simple steam generators and pumps). As part of a IAEA sponsored study it was estimated that four IRIS units have approximately the same (within 5%) capital cost as a single unit of the same total power. Construction time for the first IRIS units is expected to be 36 months, reduced to 30 after a few units experience. Construction is by pre-fabricated modules, delivered and assembled on site. Financing is obviously reduced for the smaller size, quicker construction IRIS units. Cash flow improves greatly as one unit generates power while the follower is under construction.

Advanced design, simplicity, enhanced safety, cogeneration, ease of operation, affordable cost are all characteristics which make IRIS very attractive for countries (especially smaller size ones) approaching nuclear power for the first time. It must be emphasized that IRIS is not a “paper” design. About 90% of its systems and components are the same or very similar to AP1000 and operating Westinghouse PWRs. The remaining 10% will be exhaustively tested as part of the NRC Design Certification process.

An unique IRIS feature is its organizational structure. Westinghouse leads a team of 24 organizations from 10 nations, which includes industry, laboratories, universities, government organizations and power producers. While the team’s industrial members will contribute to build up the technical infrastructure, the team’s laboratories, universities and government organizations will help in building the institutional and human infrastructure so necessary for newcomer countries to launch a nuclear program.

With sustained development and resources, IRIS expects to achieve NRC Final Design Approval around 2015 and be operational about 5 years later.

Typical Technology of Mechanics on Gen-III Passive NPPs and Gen-IV Advanced Supercritical Light Water Reactors

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Technical requirements for Gen-III advanced nuclear power plants, which take passive reactors as the main body, were originally brought forward in American "Advanced Light Water Reactor Utility Requirement Document" (ALWR-URD) in early 1990's. The primary characteristic of passive nuclear power plant is large amount of simplification to the original active safety systems, replacing or supplementing them with passive safety systems, which enhances safety and economy. However, the replacement of active safety systems by passive safety systems also brings about some mechanics that compel attention, typically, such as load-carrying capability evaluation for steel containment, in-vessel retention (IVR) of molten core debris, seismic design without OBE, thermo-hydraulic issues concerning with coupling between two-phase fluid and solid, etc.

At the beginning of this century, six typical Gen-IV advanced reactor types (Sodium Cooled Fast Reactor, Supercritical Water-Cooled Reactor, etc.) were put forward. Among these types of reactors, Supercritical Water-Cooled Reactor adopts supercritical water as coolant and operates above the thermodynamic critical point of water by increasing temperature and pressure of the coolant, which makes the plant economic and efficient. However, this type of reactor also brings about some mechanical difficulties (e.g. pressure fluctuation caused by the supercritical fluid in the core, creep of materials working at high temperature, etc.) for the design of facility and components.

In this paper, the issues mentioned above are outlined for further consideration.

Study on Thermal-Hydraulic Characteristics of Supercritical Water Reactor

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As the only water-cooled reactor among the six under the Generation-IV program, supercritical water-cooled reactor (SCWR) has its special characteristics, and takes up attentions extensively all over the world. Many countries have proposed the concept design of SCWR, such as SCLWR (Supercritical LWR) in Japan, HPLWR (High Performance LWR) in Germany, Candu-SCWR in Canada. Due to a rapid variation of the thermal-physical properties near the pseudo-critical line, the thermal-hydraulics characteristics of water at supercritical conditions differ obviously from that at sub-critical conditions of PWR and BWR. Simultaneously, because of high temperature, avoiding excessive hot spots of local cladding temperature becomes a big challenge in SCWR [1]. Therefore, sub-channel analysis is of crucial importance in designing fuel assemblies. The development of the SCWR [2] and the study on the thermal-hydraulic characteristics of SCWR [3] have been reviewed by Shi and Zhao.

Based on the PWR subchannel analysis code of SNERDI (Shanghai Nuclear Engineering Research and Design Institute), a subchannel code for SCWR was developed in this paper. Using the developed subchannel code, the thermal-hydraulic characteristics of the typical SCWR fuel assembly with the moderator water rod, including the temperature, mass flux, fuel rod cladding temperature, heat transfer coefficients and so on, is investigated.

The reference fuel assembly design is typical square configuration, with 36 moderator water rods and 301 fuel pins. In order to achieve a sufficiently high moderation ratio, the feed water entering the pressure vessel is divided into two parts. One part flows through the moderator water rods downward, and the other goes through the dowcomer to the bottom of the reactor core, where it merges with the moderator flow. Because of symmetry, 1/4 assembly is analyzed in present. Two different radial power distributions were considered according to neutronic evaluation. One was single fuel rod enrichment, and the other was an ideal enrichment distribution that would make the power in all the fuel rods the same. A symmetric chopped cosine axial power profile with a peaking factor of 1.55 was used for all the fuel rods in the assembly.

Typical outlet temperature distribution for the single enrichment with 50% flow through the moderator water rods is illustrated in FIG. 1. The temperature profile closely follows the assembly power distribution. The highest outlet temperature appears in channels 2 and 14, which are located near the center of the fuel assembly. As for the uniform power distribution, the temperature distribution, which the highest outlet temperature is at the out corner of the fuel assembly, is more uniform than the single enrichment case.

FIG. 2 shows the averaged coolant temperature and density profiles along the flow direction for different percents of flow through moderator water rods. The percents of flow through moderator water rods were varied from 0% to 75%. The results show that along the axial of the core, the coolant temperature firstly increases rapidly, forms a plateau because of very high thermal capacity near the pseudo-critical line, and then increases again rapidly with low thermal capacity of water. But the coolant density decrease sharply in the vicinity of the pseudo-critical line. For different percents of flow through the moderator water rods, the heat transfer from coolant channels to moderator water rods is different, which further induce the different inlet temperatures of coolant channels. With an increase of the percent of flow through moderator water rods, the coolant temperature rise and the density decrease obviously. However, because of total energy equilibrium, the outlet temperatures are the same.

All results above are obtained using Bishop' supercritical water heat transfer correlation [4]. A sensitivity study of different supercritical water heat transfer correlations is also performed on the thermal-hydraulic characteristics. The results show that obvious difference occurs when using different supercritical water heat transfer correlations. Further experimental work should be carried out to investigate which correlation is more compatible with SCWR core flow conditions.

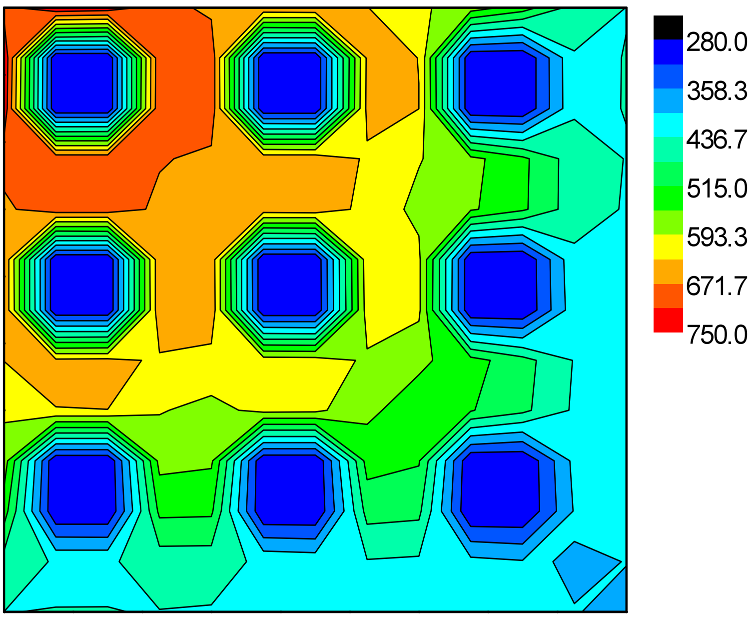
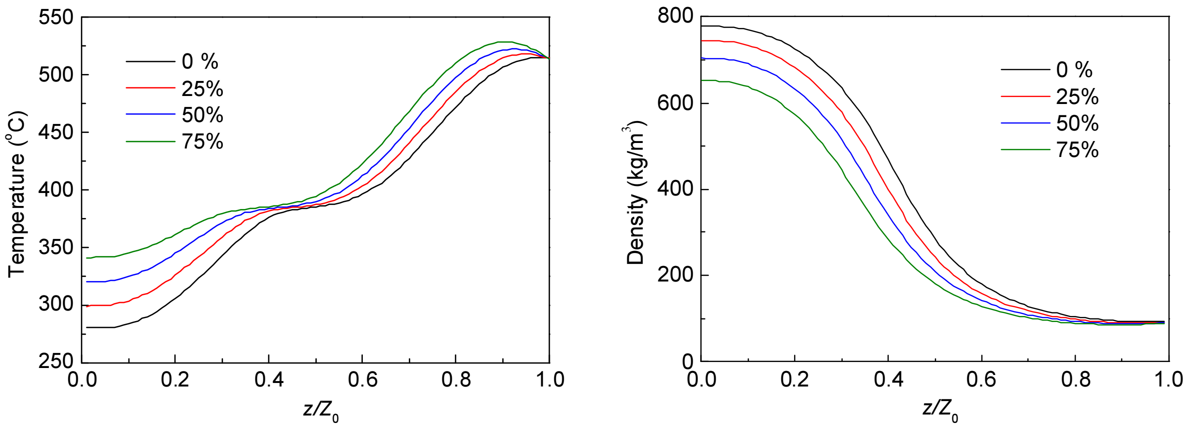


FIG. 1. Typical outlet temperature distribution of 1/4 fuel assembly for single enrichment with 50% flow through moderator water rods.



(a) (b)

FIG. 2. Typical thermal-hydraulic characteristic profiles for different percent of flow through moderator water rods: (a) coolant temperature; (b) coolant density.

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Optimization of Thorium Fuel Application Through A Symbiotic System Consists of Large FBR and Small Water Cooled Satellite-Reactors

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Global energy demand is continuously increasing over years due to the economic and social human activities development. So far, fossil fuels are projected to continuously supply a majority of the energy demand [1]. As a consequence, the supply has affected on decreasing environmental quality in term of climate change and global warming due to CO_2 emission. Therefore, utilization of clean energy and of energy sources which provides high energy released per unit mass should be introduced to give a high power production and provide an affordable price for the public. For those purposes, nuclear energy source is one of reliable options.

Another fact, developed and developing countries currently undergo different situation concerning to demand and supply of energy. The former encounter a relatively small growth of demand and have a good capability of supply due to their capital and advanced technological capabilities, therefore, they are capable to develop and operate large scale nuclear reactors, and other advanced nuclear fuel cycle facilities. The latter, in contrary, encounter a great increase of demand due to the great population growth but they have limited capabilities of supply due to the less capital and technological capabilities. Therefore, it becomes a good reason to bring up a proposal of collaboration among countries to achieve a sustainable nuclear energy utilization and development.

Considering the significant role of the small simple long life reactors for sustainable energy utilization in developing countries [2] and of the capital and technological capabilities of developed countries, there is a strong need for realization of a symbiotic system consisting of large reactors operated in developed countries and small satellite-reactors, *SSR*s, deployed in developing countries. The basic idea of symbiotic nuclear reactor system is how different scales and types of nuclear reactors are possible to be operated and can provide mutual benefits each other. In the present study, a symbiotic system, as shown in FIG. 1, between large fast reactors and small thermal reactors is proposed with different main targets of the small and large reactors.

In the proposed system, a symbiotic system of large power scale 3000 MWth fast breeder reactors (*FBR*s) and small-satellite long life water cooled thorium reactors (Small-Satellite WTRs) as *SSR*s is investigated. The FBR is designed for generating electricity and at the same time producing ^{233}U. To produce ^{233}U, natural thorium fuel pins are introduced together with natural uranium pins in the FBR mixture core. The SSR, WTRs (30 ~ 300 MWth) with thorium fuel, are designed for supplying small scale of energy demand for less-developed areas of developing countries.

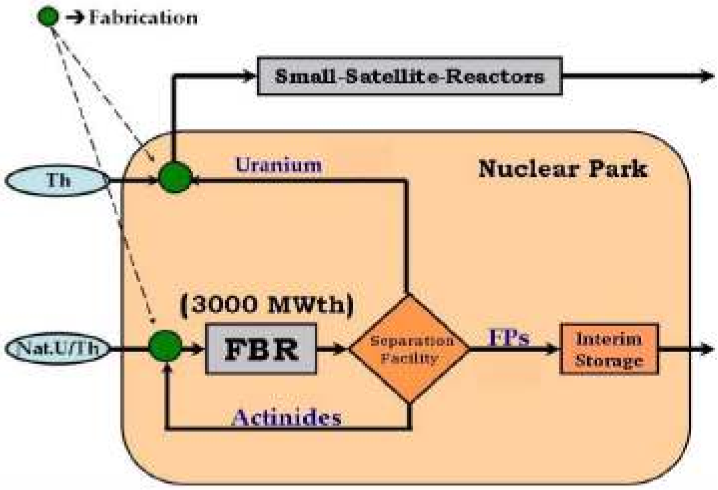

FIG. 1. Fuel cycle scenario of the proposed symbiotic system

The basic reasons for employing the two types of reactors (large FBR and small WTRs) are as follows:

a) Water cooled reactors are chosen as basic technology of *SSR*s because they are based on the current thermal reactor technology so that they are more feasible to be operated in areas which have limited infrastructures and resources. Another point is that ^{233}U is superior to other fissile materials in thermal reactors, therefore a better burnup performances of the SSRs can be expected. But no thermal reactors can be critical if fueled by natural thorium only and it should be enriched by fissile material [3]. In thorium case, ^{233}U is the best fissile but it does not naturally exist. Therefore, ^{233}U should be provided by other nuclear reactors which have capability to produce it.

b) A metallic-fueled, sodium-cooled FBR is used for producing ^{233}U fissile material because fast reactors provide better neutron economy compared to thermal reactors and high conversion ratio can be achieved. The high conversion ratio will provide high ^{233}U fissile discharge rate from the FBR core. Sodium-cooled FBR is the current advanced technology and metallic-fuel type has been investigated that achieves higher fissile production than oxide-fuel type for sodium-cooled FBR [4].

From the present study, it can be expected that a great number of small scale energy demands in developing countries can be significantly supplied by applying the proposed symbiotic systems.

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Pressure Tube Reactor Contribution to Sustainable and Secure Energy

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It is well known that electricity use and economic health are coupled. Major economic indicators (GDP/capita) are therefore highly correlated with kWh/capita in both developing and developed industrial nations. It can be shown that at least 4000 new reactors. some ten times the present number, would be needed to assist the contribution of non-carbon energy sources to stabilizing greenhouse gas (GHG) concentrations at ~550 ppm by 2050-2100, and also enable fuel switching from hydrocarbons in transportation. This is in addition to requiring massive investment in wind power facilities, and extensive conservation and efficiency measures.

Nuclear power program decisions in many nations will be increasingly based on political, strategic and economic considerations involving the complete nuclear fuel cycle, including resource utilization, radioactive waste disposal, proliferation resistance, and supply assurances. The overall direction of development must and will provide an integrated and complementary reactor and fuel cycle, including the deployment of alternative fuel enrichment and reprocessing technology and services. The full fuel cycle still has negligible GHG emissions.

We show how nuclear energy can meet the inexorably growing demand, without increasing costs and avoiding the issue of security of energy supply and environmental emissions of carbon dioxide and othcr GHGs. This paper examines and places in context all these aspects via the viewpoint of developing a sustainable global nuclear fuel cycle, and proposes alternate paths to peace, prosperity and non-proliferation that are somewhat outside the present rather traditional thinking. They still represent existing and known technology opportunities that may run counter to many current national positions, and today’s commercial and technical interests, while presenting very large opportunities.

We discuss how the global nuclear fuel cycle must support and maintain international trade, and address energy and environment needs; how optimizing nuclear power, and associated fuel cycle and waste management technology ensures economic and environmental sustainability; why developing alternate “closable” thorium fuel cycles (enrichment, reprocessing, separation and advanced cycles) can meet the needs; and argue that nuclear power technology remains competitive and contributes to national and international energy supply and security, while addressing proliferation risks.

The Modernization Program, Power Uprate at NPP V2 Jaslovske Bohunice

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SE a.s.-Enel own an optimal production portfolio comprised of **nuclear, thermal and hydroelectric** power plants

As concerns V2, the upgrading is based on three main pillars:

- the modernization
- the power up-rate
- the ageing monitoring program

The main targets of the modernization project are:

- The increasing of the Nuclear Safety and of the Nuclear operational reliability
- And the seismic improvement

obviously in full compliance with IAEA requirements and with the decisions by the Nuclear Regulatory Agency of the Slovak Republic (UJD) and achievement of the probabilistic safety criteria in accordance with IAEA recommendations. And, last but not least, the possibility to ensure a safe, reliable, economical and effective electricity and heat generation

Management of SLOVENSKÉ ELEKTRÁRNE approved in 1997 the following plan of activities for V2 MODERNIZATION:

- Refurbishment of the Mechanical emergency and safety systems: Pressurizer, Main Cooling pumps, Primary circuit, Steam generators, Emergency core cooling, Residual heat removal, Hermetic zone, Super emergency feed water, Technical essential water, Main steam and feed water piping, Ventilation,
- Refurbishment of the Electric emergency and safety systems: Sectional and secondary switchboard cabinets, Electric own consumption supply, Automatics on the primary and secondary circuit, Diesel generators, Accumulator batteries, Reserve electric supply.
- Refurbishment of the Instrumentation and Control emergency and safety systems: Reactor trip, Reactor power control, Reactor power limitation, Neutron flux, Breakers, Engineered Safeguards Features Actuation Systems, Sensors, Information Computer systems, Post Accident Monitoring system, Radiological monitoring.
- Seismic improvement, Fire protection, Components qualification and classification
- Safety documentation

POWER UP-RATE:

- Increase of secondary equipment efficiency
- Increase of Reactor thermal power up to 107 %

Actions planned on primary and secondary systems to create the technical conditions for power uprate:

- Implementation of calibration systems for a more accurate reactor power control in terms of temperature and neutron flux
- Upgrade of turbines**:** Modifications of High Pressure and Low Pressure part. We are going to replace most of the inner components of the original equipment
- Main condensers : General arrangement was changed into a modular, complete retubing was performed, water chambers were reinforced.
- Cooling towers : All 4 cooling towers are nowadays in operation after replacement of the cooling system.
- Generators : The complete scope of work is the following:
  - Modification of Stator
  - Modification of Rotor
  - Replacement of Accessories
- Unit transformer
  - Replacement of winding
  - Replacement of bushings
  - Replacement of cooling system
  - New diagnostics system
- Connections to the grid
  - Modification of encapsulated wires
- Modification of power output connections to generator and to transformer
  - Replacement of generator disconnectors
- Instrumentation and control system
  - Replacement of turbine and generator protection systems
  - Replacement of feed water flow meters in the SG to increase the precision of the measurement of reactor power output
  - Replacement of steam flow-rate sensors and installation of moisture separators in the secondary circuit
  - Replacement and functional enlargement of automatic regulation system of turbine (the old system was hydraulic, the new one will be electronic)
  - Upgrading of instrumentation and automatic controllers both in the primary and secondary circuit

Safety and Performance Achievement in Czech NPPs with VVER Reactors

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The paper will provide an overview on the performance achievements of the Czech NPP with VVER reactors, in particular the NPP Dukovany with 4 units of VVER 440 and the NPP Temelin with 2 units of VVER 1000. A principal decision on the change of the nuclear programme from HWGCR to PWR (VVER) was made in former Czechoslovakia in 1970. NPP Dukovany was put in operation between the years 1985 and 1987, thus its first unit is approaching the age of 25 years. NPP Temelin is rather younger; its two units were put in operation in 2000 and 2002. All together these units have produced so far 370 TWh of electricity (approx. 6x annual consumption) and accumulated more than 100 reactor-years of operating experience. The paper will provide basic statistics on operational history of these two NPPs, including major safety related events, operational feedback and lessons learned.

Since the real beginning of their operation all units have been subject of continuous performance improvements and safety enhancements based in particular on the IAEA recommendations resulting from the "Extrabudgetary programme" on safety upgrading of VVER reactors (safety issues). As already documented in the National Report under Nuclear Safety Convention all safety issues had been successfully addressed. As a part of safety upgrading programmes (Back-fitting, Morava) there were implemented more than 130 safety measures in NPP Dukovany and 90 in NPP Temelin. The paper will briefly characterise these measures with a conclusion that the safety of all these units have been significantly enhanced and that their level of safety is comparable with western PWR units of the similar age. These results have been verified and confirmed in a number of safety review missions of the IAEA (OSART), WANO, including a special review carried out in 2004 in the process of accession of the Czech Republic into European Union.

At present the operation of NPP Dukovany is fully stabilized reaching annual load factors around 90%. The units meet basic safety requirements of the IAEA and WANO and in certain areas, such as annual doses of operating and service personal they reach unique values. Recently a new advanced fuel have been implemented in all four units of NPP Dukovany with the aim of prolonging the fuel cycle up to 5 years with significantly higher burnup (65 MWd/kg for fuel pin). This will significantly improve the utilization of fuel and contribute to better economy of the NPP operation.

With the progressing age the NPP Dukovany undergoes since 2002 through a substantial refurbishment and equipment retrofit aimed in power uprating by more than 11 %, including replacement of the origin analogue I&C systems by the digital one. The paper will discuss main conclusions of the recent study on the feasibility of extending the operation lifetime up to 60 years.

After certain “birth pain” the NPP Temelin has also reached a stabilized operation although its load factors are rather lower compared to NPP Dukovany. Ongoing long term problems with the integrity of fuel have lead to a principal decision to replace the supplier of fuel from Westinghouse to TVEL. This “come back” to Russian fuel is a challenge to all involved parties – fuel supplier (TVEL), the utility (CEZ plc) and the Czech safety authority (SUJB) as the licensing of the new fuel will have to demonstrate its higher level of safety and reliability.

In the operation of all Czech units modern tools and methods based on best estimate computer codes and living PSA studies with their practical applications such as risk monitor, risk informed ISI, etc. are intensively used with the help of the Nuclear Research Institute Rez plc which traditionally continues to play the role of the key technical support organization for the safe operation of the Czech NPPs. The paper will briefly inform on NRI’s efforts to maintain its competence and expertise by active participating in international programmes and projects of the IAEA, OECD/NEA and EURATOM, including bilateral cooperation.

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Harmonization Program Actual program of increasing the safety of The Dukovany Nuclear Power Station is included in the newly outlined Harmonization Program. This program is not only focused on questions connected with the power station project change and replacement of some equipment but also covers other fields that can effect safety of the power station. The highest importance or contribution to increasing of safety need not to be, as generally assumed, the improvements of the facilities. A higher level of safety can be achieved by improving the safety culture, too. The Dukovany Nuclear Power Station's target is to achieve, by implementation of the Harmonization Program, a reduction of the reactor core damage probability coefficient from the present value of $1.7*10^{-5}$ to value of $7.7*10^{-6}$ in the year 2010 (this value means that an event causing fuel damage in the reactor core can occur probably once in the period of 130,000 years). This value has been recommended for newly built power stations by the Atomic Energy International Agency.

Safety and Performance Achievement of Indian Nuclear Power Plant

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The Nuclear power programme in India at present is based mainly on series of pressurized heavy water reactor(PHWR). Starting from Rajesthan atomic power station comprising two unit of 200 MWe in 1973, the programme has come a long way with 17 units in operation includes two unit of 540 MWe at Turapur 3&4. Narora atomic power station commissioned in 1991 marked major indigenization and standardization of PHWR design. Which includes, double containment of reactor building, two diverse reactor protection system and emergency core cooling system. Further in 540 MWe at Tarapur, the safety systems have been divided into two groups to the extent possible. These groups are physically seperated so that any common mode incident either inside or outside the reactor building would nor disabled more than one of these groups. Each group of safety system should meet the requirements of shutdown the reactor , remove decay heat from the fuel subsequent to shutdown, prevent any subsequent escalation of failures, minimize the escape of radioactivity, supply necessary information to the operators for assesment of the state of the plant. Group-1 is first line of defense safety systems i.e. Shur Down System-1 (SDS-1), Emergency Core Cooling System (ECCS) and all process water systems including shutdown cooling. Group-2 is the second safety systems i.e. Shut Down System-2 (SDS-2), containment isolation, Moderator cooling, Emergency water supply (fire fighting water with diesel driven pump) through Steam Generator. Status of the plant is monitored and controlled from Main Control Room (MCR) and it is done from supplementary control room in case of emergency. Each safety system is designed to achieve unavailibility of 10^{-3} year/year or less.

Indian nuclear power plant progressively attained excellent operation performance comparable to international benchmark with sustained effors in operation and maintenance, development and nurturing of qualified human resources at various level, development of various quality equipment and full scope simulator facility at new plant. In the year 2002, Kakrapar Atomic Power Station (KAPS) was adjudged the worlds' best operating nuclear plant among the PHWRs world over. Kaiga-2 completed 529 days of continuous operation recently. Prior to Kaiga-2, many Indian reactors KAPS-1, NAPS-2, RAPS-4 also registered continuous operation of nearly a year. The biennial shutdown duration have also been reduced to about 20 days by effective planning and management of shutdown activities. NPCIL units are operating consistently at high availability factor of about 90%.

The safety track record of 294 reactor years of operation evidences the strong commitment to safety and demonstrates strong safety culture of Indian nuclear industry. In India nuclear power plants, the radiation dose to occupational workers has been a small fraction of the limits prescribed by Atomic Energy Regulatory Board (AERB), the regulatory authority. The dose to environmnent bas been insignificant fraction of the prescribe limit. The limit prescribed by the AERB are more conservative than the international limit set by the International Commision on Radiation Protection (ICRP).

In the earlier design of PHWRs used at RAPS-1&2, MAPS-1&2, NAPS-1&2 and KAPS-1 a zirconium alloy (zircalloy-2) was used for coolant channel. It was considered best available material at that time. However, in pile of experience brought out, the requirement of replacement of coolant tubes after 10 to 12 effective full power years in view of the modification of material characteristics especially the reduction in mechanical strength due to to hydriding under radiation during service. These coolant channels of RAPS-2, MAPS-1&2 and NAPS-1 were replaced in record time by zirconiuin-2.5% niobium and NAPS-2 & KAPS-1 En masse coolant channel replacement is in progress. Although in Indian PHWR, no significant thinning was observed in feeder pipe and elbow, but as a precautionary measure, en masse feeder replacement were carried out at RAPS-2, MAPS 1&2 and NAPS-1. The long shutdown of EMCCR/EMFR were used to carry out safety upgrades in parallel. These units were brought to the state of the art in terms of safety. Several major equipment, like the steam generators in MAPS l&2 were also replaced to enhance the life of the plant and performance of equipment during these upgrades.

Tarapur unit 1&2, commisioned in the year 1969 were licenced to operate for 25 years. NPCBL developed and carried out probabilistic safety analysis (PSA) of TAPS l&2, which found that the reactor were operating in the safe regime and met the current standards. However, based on the PSA studies and ageing management programme, system upgradation and siesmic upgradation were carried out in 2005/2006. These includes replacement of 3x50% capacity emergency diesel generators (DGs) with 3x100% capacity DGs, introduction of supplementary control rooms, upgradation of electrical, fire protection, ventillation and instrumentation and control systems. Siesmic strengthening of civil structures safety systems, pipping end equipment was carried out by providing additional anchorage and supports.