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Innovative small and medium sized reactors: Design features, safety approaches and R&D trends

Final report of a technical meeting held in Vienna, 7–11 June 2004

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FOREWORD

There is a renewed interest in Member States in the development and application of small and medium sized reactors (SMRs). These reactors are most suitable for deployment in the developing countries with low electrical grid capacity and in countries with low electricity demand projections. SMRs are also the preferred option for non-electrical applications of nuclear energy such as desalination of seawater, district heating, hydrogen production and other process heat applications. In the past, the trend in nuclear power reactor technology development showed an emphasis towards large reactors due to the economies of scale, which produced reactor designs on to 1600 MWe. A development of SMRs points into the opposite direction, i.e. towards smaller outputs with an equivalent electrical power of less than 700 MWe.

In order to beat the economy of scale SMRs have to incorporate specific design features that result into simplification of the overall plant design, modularization and mass production. Several approaches are being under development and consideration, including the increased use of passive features for reactivity control and reactor shut down, decay heat removal and core cooling, and reliance on the increased margin to fuel failure achieved through the use of advanced high-temperature fuel forms and structural materials. Some SMRs also offer the possibility of very long core lifetimes with burnable absorbers or high conversion ratio in the core. These reactors incorporate increased proliferation resistance and may offer a very attractive solution for the implementation of adequate safeguards in a scenario of global deployment of nuclear power.

The activities on design and technology development for SMRs are ongoing in many countries, and there are growing expectations of an increased support from the IAEA to interested Member States in the definition of common technology and infrastructure development needs and in the coordination of selected international R&D efforts for such reactors. Reflecting on this demand, on 7-11 June 2004 the IAEA convened a Technical Meeting on Innovative Small and Medium Sized Reactors: Design Features, Safety Approaches and R&D Trends, which was attended by 15 experts from 12 Member States. The presentations and discussions at the meeting addressed about 30 concepts and designs of innovative SMRs and several options for the innovative nuclear energy systems on their basis.

This publication has been prepared through the collaboration of all participants of this meeting and presents its final report, which summarizes the major features and identifies the technology and infrastructure development needs common to certain groups of the SMR concepts and designs considered at the meeting.

The IAEA appreciates the support of all participants and authors who provided inputs and assisted in the preparation of this TECDOC. Especially appreciated is the contribution of D.C. Wade (United States of America) who was a chairman of this meeting.

The IAEA officer responsible for this publication was V. Kuznetsov of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

A development of small and medium sized reactors (SMRs)[1] is supported by the following major arguments:

- The principal drivers behind the projected large increase in global energy needs are population growth and economic development in today's developing countries [1], which often have insufficient infrastructure and small electricity grids. The reactor fitting into a SMR range may be a good choice to meet the demand of such countries;
- Many developing countries have limited investment capability, especially as comes to funds in hard currency. In this context, SMRs may become the only affordable nuclear power option for such countries;
- In industrialized countries, the electricity market deregulation is calling for a flexibility of power generation and applications that smaller reactors may offer. In particular, the SMRs of modular design provide for an incremental capacity increase, which makes it possible to spread the investments in time and to reduce the associated financial risk;
- SMRs are of particular interest for both near-term, e.g. seawater desalination, and advanced future non-electrical applications, such as hydrogen production, coal liquefaction, and other process heat applications;
- New technologies cannot be deployed at once to a large scale. Learning from a small prototype plant may be necessary to reach the final goal of their wide-scale deployment.

About 50 concepts and designs of the innovative SMRs are under development in more than 15 IAEA Member States representing both industrialized and developing countries. SMRs are under development for all principle reactor lines, i.e., water cooled, liquid metal cooled, gas cooled, and molten salt cooled reactors, as well as for some non-conventional combinations thereof.

Upon a diversity of the conceptual and design approaches to SMRs, it may be useful to identify the so-called enabling technologies that are common to certain reactor types or lines. An enabling technology is the technology that needs to be developed and demonstrated to make a certain reactor concept viable. When a certain technology is common to several SMR concepts or designs, it could benefit from being developed on a common or shared basis. The identification of common enabling technologies could speed up the development and deployment of many SMRs by merging the efforts of their designers through an increased international cooperation.

Identification of the enabling technologies for SMRs may also facilitate a link to the national or international technology development for nuclear reactors beyond the SMR range. In turn, this will contribute to a dialogue between the major nuclear vendors and the potential national or regional users, which may help define how the developments of a few industrialized countries could be later on adjusted to the specific needs of developing countries or regions.

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[1] According to the classification adopted by the IAEA, 'small reactor' is a reactor with the equivalent electric power less than 300 MW, 'medium sized reactor' is a reactor with the equivalent electric power between 300 and 700 MW.

Apart from an option to benefit from technology development on a common basis, there are several trends of infrastructure development that may be of benefit for the deployment of many SMRs. Some targeted infrastructure changes, such as an introduction of technology-neutral safety requirements, may be of benefit to all innovative reactors, independent of their size. Other infrastructure developments, such as reestablishment of a practice of licensing by the prototype demonstration, could be of special value namely to SMRs.

Upon the advice and with the support of its Member States, the IAEA provides a forum for the exchange of information by experts and policy makers from industrialized and developing countries on the technical, economic, environmental, and infrastructure aspects of the SMR development and deployment in the 21st century [1,2]. On 7-11 June 2004 the IAEA convened a Technical Meeting on Innovative Small and Medium Sized Reactors: Design Features, Safety Approaches and R&D Trends, which had the following main objectives:

(1) To provide a forum for the exchange of information on the state-of-the-art in the development, design and demonstration of innovative[2] small and medium sized reactors (SMRs) with a focus on:

- Innovative approaches pursued to facilitate the solutions for one or several issues accepted as critical for further deployment of nuclear power;
- The enabling technologies and infrastructure development needs for SMRs;
- The application potential of SMRs, including a variety of possible non-electrical applications and special features of the SMR plants, such as modularity, transportability, lifetime core operation and factory fabrication and fuelling;
- New approaches to the implementation of inherent safety features and passive safety systems;
- Small reactors without on-site refuelling;

(2) To support the preparation of an IAEA report on the status of innovative SMR designs and other SMR-related activities by the IAEA, such as a report on small reactors without on-site refuelling and a report on the review of passive safety design options for SMRs.

This TECDOC presents a variety of innovative water cooled, gas cooled, liquid metal cooled and non-conventional SMR designs developed worldwide and examines the technology and infrastructure development needs that may be common to several concepts or lines of such reactors. The TECDOC also gives an updated definition of small reactors without on-site refuelling and provides a preliminary review of the passive safety design options for SMRs.

2. THE SCOPE OF INNOVATIVE SMR DESIGNS

Fifteen experts nominated by the IAEA Member States: Argentina, Brazil, China, India, Indonesia, Japan, France, the Republic of Korea, South Africa, the Russian Federation, the United Kingdom, and the United States of America attended the meeting, submitted papers and delivered the presentations covering about 30 designs of innovative SMRs, including:

[2] Ref. [3] defines an innovative design as the design “that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice” and would, therefore, “require substantial R&D, feasibility tests and a prototype or demonstration plant to be implemented”.

- Integral type pressurized water reactors targeted for near term deployment: SMART (the Republic of Korea), IRIS (the International Consortium, led by Westinghouse, USA), CAREM (Argentina), and SCOR (AREVA-CEA, France);
- Small pressurized water reactors without on-site refuelling from Russia: SAKHA-92, ABV-3, ABV-6, KLT-40S (with lifetime core), VBER, RIT (all from OKBM), RUTA-70, UNITHERM, NIKA-70 (from RDIPE), in particular, designed for floating NPPs;
- Direct conversion small light water reactor without on-site refuelling ELENA (RRC "Kurchatov Institute", Russia);
- Light water cooled heavy water moderated pressure tube reactor AHWR (BARC, India);
- Light water reactors using coated particle or pebble bed type fuel: PFPWR50 (University of Hokkaido, Japan), VKR-MT (VNIIAM-RRC "Kurchatov Institute", Russia), FBNR (Federal University of Rio Grande Do Sul, Brazil);
- Innovative high temperature gas cooled reactors: PBMR-400 (ESCOM, South Africa), HTR-PM (INET, China), HTR-F/VHTR (AREVA-CEA, France);
- Lead-bismuth cooled small reactor without on-site refuelling SVBR-75/100, targeted for near-term deployment (IPPE and EDO "Gidropress", Russia);
- Innovative lead or lead-bismuth cooled small reactors without on-site refuelling: STAR-LM, STAR-H2, SSTAR ("STAR family", ANL, USA), SPINNOR and VSPINNOR (ITB, Indonesia);
- Lead-bismuth cooled compact high temperature reactor CHTR, with HTGR type fuel (BARC, India);
- Molten salt cooled small reactor with pebble-bed fuel MARS (RRC "Kurchatov Institute", Russia);
- CANDLE burn-up concept for small high temperature gas cooled reactors and for small reactors with fast neutron spectrum (RLNR TITech, Japan).

3. CROSS-CUTS OF SMR DESIGNS

3.1. Timeline of readiness for deployment

Figure 1 gives a projection for the timelines when the demonstration prototypes of certain SMRs could be deployed. This projection is based on the designers' evaluation of the time needed to carry out necessary R&D and to pass the required design certification and licensing procedures, all under favourable conditions of financing. No consideration of the unequal starting conditions and, therefore, varying prospects for the attraction of investments was made. Some SMRs implement more radical innovations, and an essential modification of the existing regulations may be needed for them ever to get licensed. The projection of Fig. 1 makes no account of the time needed to develop and enforce a new set of regulations, which may be a more complicated and lengthy process than the technology development itself.

The data in Fig. 1 are exclusive responsibility of the designers of their respective SMRs. The IAEA secretariat has introduced no corrections or adjustments to these data. As an example, the authors of a fixed bed nuclear reactor (FBNR, Brazil) claim their design to be simple and thoroughly based on the existing PWR technology, which they view as a decisive factor in making it suitable for a near term deployment. However, the discussion at the meeting produced other opinions on this, pointing to a new technology for the movable pebble bed fuel that may require a considerable time to be developed, and suggesting that the deployment term for FBNR is moved into a more distant future.

3.2. Design and regulatory status of SMRs

Table 1 gives the data on the current design and regulatory status of the SMRs addressed at the meeting.

As it can be seen from both Fig. 1 and Table 1, water cooled SMRs are the most suitable candidates for a near-term deployment. The high temperature gas cooled reactors with thermal neutron spectrum follow them closely. Small PWR designs from Russia are based on the experience of the marine reactors and are said to be deployable within a very short term, once the financing for a necessary limited amount of the Research, Design & Demonstration (RD&D) becomes available.

One notable exception is the Russian lead-bismuth cooled reactor SVBR-75/100, which has reached the detailed design stage and is expected to be ready for licensing soon. The reason behind this is that the SVBR-75/100 is thoroughly based on 80 reactor-years of the operation experience of its submarine prototypes.

3.3. Evolution of designs since the last IAEA status report on SMRs

Five SMR designs presented at the meeting were previously addressed in the IAEA-TECDOC-881 Design and Development Status of Small and Medium Reactor Systems (1995) [1]. Table 2 illustrates the changes in their design and regulatory status over the period from 1995 to 2004.

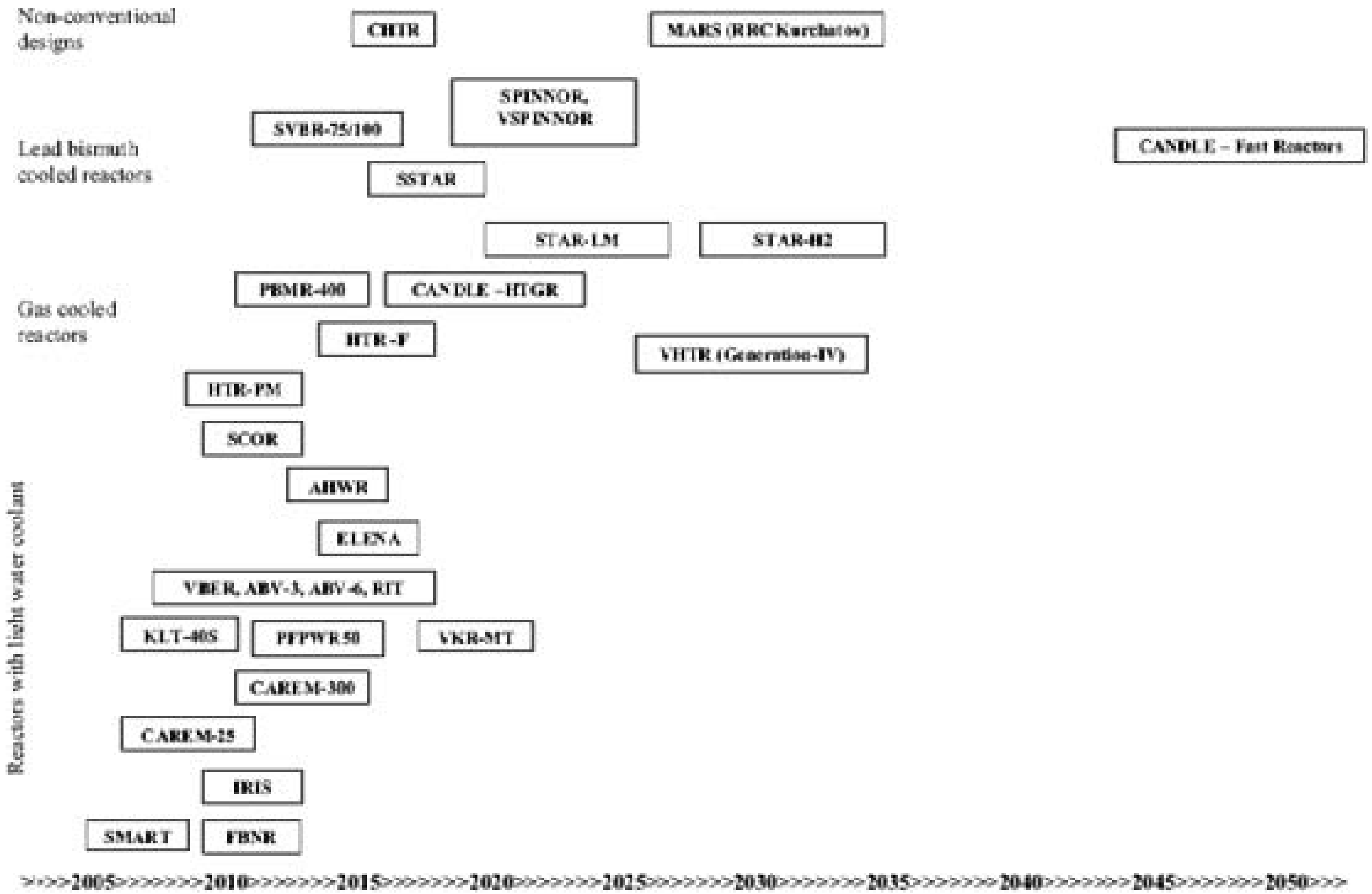

FIG.1. Timeline of readiness for deployment (projections by authors/designers of SMRs).

TABLE 1. DESIGN AND REGULATORY STATUS* OF SMRs

Licensing status	No formal licensing process	Formal preliminary licensing process	Full licensing process
Detailed designs	SVBR-75/100	CAREM-25 AHWR ABV-3 ABV-6	KLT-40S** SMART
Preliminary designs	PBMR-400 KLT-40S (lifetime core) VBER-150 RIT-150	IRIS SAKHA-92	
Conceptual designs	HTR-PM ELENA NIKA-70 UNITHERM RUTA-70 SCOR SSTAR STAR-LM STAR-H2 VKR-MT PFPWR50 FBNR SSPINOR CHTR HTR-F VHTR (Generation-IV) MARS(RRC "Kurchatov Institute") CANDLE		

* Time stage within the cell: more to the left means: at an early development stage; more to the right means at a final development stage.

** A license for the floating NPP construction at Severodvinsk (Russia) was obtained.

Though being based on a limited number of inputs, Table 2 points to a particular progress in the development status achieved by some SMR designs, such as CAREM, AHWR, KLT-40, SAKHA-92, and ABV. It could be mentioned that huge delays in the development of SMR projects, which are with no exception observed for all designs, may in some cases stimulate the designers to introduce particular modifications, for example, to focus on the reactors of different unit power (ABV, RUTA).

Other SMR concepts and designs presented at the meeting were not addressed in the IAEA-TECDOC-881 [1], which points to a significant change in the scope of SMR projects over the past decade.

TABLE 2. EVOLUTION OF THE DESIGN STATUS OF SMRs

SMR design	Design status / targeted deployment date [1]	Design status / deployment date as targeted in 2004
CAREM (Argentina)	Preliminary design/ 1995	Detailed design / 27 MW(e) prototype in 2006
KLT-40 (OKBM, Russian Federation)	Detailed design / ready for construction	KLT-40S detailed design / 2007 KLT-40S with long life core - Preliminary design/ 2009
ABV (OKBM, Russian Federation)	Preliminary design / licensing started	Detailed design (ABV-3, ABV-6) / 2008
SAKHA-92 (OKBM, Russian Federation)	Conceptual design / No data	Detailed design / 2008
RUTA (RDIPE, Russian Federation)	RUTA-20 conceptual design / no data	Ruta-70 conceptual design / no data
AHWR* (BARC, India)	Preliminary design / no data	Detailed design, peer review completed / 2012

* The Indian AHWR was not described in the IAEA-TECDOC-881.

3.4. Cross-cut of SMR applications

Table 3 illustrates a variety of SMR applications and cogeneration options. For some designs, such as HTR-PM, various co-generation options will probably be added at further stages of their design development. Many designs provide for a flexibility of different applications or cogeneration options. Such are the PWR type small reactors from Russia and the “STAR family” of lead or lead-bismuth cooled reactors from the USA. The remarkable examples of a multiple cogeneration option are provided by the PBMR-400 and by some “members” of the “STAR family”. Here, the approach is to produce electricity, hydrogen and potable water within a single balance of plant. Finally, many designers make a provision for the purposeful use of the rejected heat, which is viewed as an important factor contributing to the competitiveness of their SMRs.

TABLE 3. SUMMARY OF SMR APPLICATIONS

Application	SMR design
Electricity generation	IRIS, CAREM, SVBR-75/100, AHWR, HTR-PM, PBMR-400, VBER, SCOR, FBNR, HTR-F
Potable water production	AHWR, SSTAR, HTR-F, FBNR
District heating	ABV-3, RUTA-70, PFPWR50, FBNR
Hydrogen production*	PBMR-400, STAR-H2, VHTR (Generation-IV), CHTR, HTR-F
Process heat applications	VHTR (Generation-IV), HTR-F, CHTR
Hydrogen and potable water production	STAR-H2
Use of the rejected heat, e.g. for seawater desalination or district heating	PBMR-400, AHWR, "STAR family", ELENA, UNITHERM, ABV-3, ABV-6, KLT-40S, SVBR-75/100, VBER, RIT, MARS, HTR-F, FBNR
Co-generation of electricity with:	
• Potable water production	SMART, IRIS, CAREM-125, PBMR-400, STAR-LM, ABV-3, ABV-6, KLT-40S, SVBR-75/100, VBER, RIT, MARS, NIKA-70, SPINNOR, VSPINNOR, HTR-F
• District heating	SAKHA-92, ELENA, ABV-3, ABV-6, KLT-40S, SVBR-75/100, VBER, RIT, MARS, VKR-MT, UNITHERM, IRIS
• Hydrogen production	PBMR-400, HTR-F
• Other process heat applications (with the authors giving no details)	MARS, FBNR

* The entries in this row are the SMR concepts and designs that provide for hydrogen production through a direct use of heat for the thermochemical processes

3.5. Cross-cut of SMR special features

Table 4 presents a cross cut of special features offered by the SMRs. Out of the thirty SMR concepts and designs addressed at the meeting, about 50% rely on a modular approach providing for a flexible plant capacity, about 60% fit under the definition of small reactors without on-site refuelling, and about 35% are the factory fabricated and fuelled reactors. At the moment, nearly all barge-mounted designs come from the Russian Federation.

TABLE 4. SPECIAL FEATURES OF SMRs

Special feature	SMR designs	Comments
Flexible plant capacity achieved through a modular design	PBMR-400, IRIS, CAREM, SMART, ABV, VBER, SVBR-75/100, UNITHERM, FBNR, HTR-PM, “STAR family”, FBNR	
Small reactors without on-site refuelling	“STAR family”, ELENA, RUTA-70, MARS (Kurchatov), VKR-MT, SAKHA-92, ABV, KLT-40S with long-life core, VBER, RIT, SVBR-75/100, UNITHERM, NIKA-70, FBNR, PFPWR50, SPINNOR/VSPINNOR, CHTR, options of long-life core for IRIS and CAREM, long-life core by CANDLE	The term “refuelling” could be defined as the removal and/or replacement of either fresh or spent, single or multiple, bare or inadequately confined nuclear fuel cluster(s) or fuel element(s) contained in the core of a nuclear reactor. Such definition allows for the inclusion in this category of those small reactors that are designed for an infrequent replacement of well-contained fuel cassette(s) in a manner that prohibits a clandestine diversion of nuclear fuel material. It also includes factory fabricated and fuelled reactors in a generic way
Factory fabricated and fuelled reactors	SAKHA-92, ABV, KLT-40S with long-life core, VBER, RIT, SVBR-75/100, UNITHERM, NIKA-70, FBNR, SPINNOR/VSPINNOR, CHTR	The transportation of assembled and fuelled reactors from the factory to a site and back is required after each reactor lifetime or after each fuel lifetime*
Floating, e.g. barge-mounted NPPs	ABV, KLT-40S, VBER, RIT, NIKA-70, IRIS, SVBR-75/100**	As an alternative, the projects of land based NPPs are being developed for the ABV, VBER, and IRIS

* Some designs, such as SAKHA-92, VBER and RIT provide for their transportation from a site back to the factory after each fuel lifetime. For them, factory refuelling is combined with the equipment repair.

** A floating NPP option is being developed for the SVBR-75/100.

4. MAJOR FINDINGS

4.1. Diversity of approach, missions and time frames

The presentations and discussions at the meeting reflected multiple activities on-going for the innovative SMRs in many Member States. They covered a variety of SMR design and safety approaches targeted at a near-, medium- and longer-term deployment. They also covered the expanded range of energy products, including hydrogen, potable water, process and residential heat. All concepts addressed the perceived need to provide energy services at a low initial buy-in cost or leasing rent.

SMRs have many common issues related to the provision of high economic competitiveness, enhanced safety, and adequate proliferation resistance. Innovative approaches are needed to resolve these issues, and finding a solution to many of them could benefit from an increased international cooperation. The deployment opportunities for SMRs could be increased with a promotion of certain developments in legal, institutional, infrastructure, and public acceptance areas. Among them: achieving the reciprocity of licensing agreements, securing the insurance of fuel supply, and finding a pathway to simplified licensing procedures, e.g. the reduced or eliminated off-site emergency planning and/or simplified licensing of a replicate plant construction. The presentations clearly indicated that, for different countries and regions, different solutions for nuclear energy systems with SMRs could be preferable, as defined by different national or regional constraints. Several innovative approaches for the deployment of SMRs were outlined, such as Build-Operate-Transfer (BOT), e.g. realized with the floating NPPs or factory fabricated battery-type reactors.

The need to break an economy of scale was clearly identified as an objective of prime importance for all SMRs. Several factors arising from the on-going liberalization of energy markets were mentioned as being particularly in favour of the NPPs with SMRs, among them: the economy of multiple small modules and the associated financial risk reduction; an option to spread the investment costs in time by applying a modular approach to the NPP design; the diversity of SMR designs, capacities, and applications as a factor of merit in liberalized markets; and the flexibility of an SMR-based energy system not only as a desirable feature but as a principle requirement to such systems. A presentation on the IRIS put it in short as "SMRs do not benefit from the economy of scale, but they can have an economy of multiples. To achieve this, worldwide market is needed, and one option to expand to a worldwide market is to increase international cooperation".

The insight of a presentation from Indonesia was that 93% of the electricity load in this country, and perhaps in some other developing countries, is residential. Therefore, such countries may try to attract the industry to use an off-peak load and/or to use the power plants that are capable of a load follow operation. The same presentation outlined that some communities may prefer to simply buy the electricity and leave everything else to the provider. The portable SMR concepts from the OKBM, Russia might well meet such a demand. The example of an international approach targeted at improving the deployment perspectives for an innovative SMR was given in a presentation on the IRIS project. This project also provides a good example of cooperation between industry, research institutions, and academia. Other examples of such cooperation at a national level were given in the presentations on SMART, CAREM, HTR-PM, PBMR, KLT-40S and other SMRs.

A summary of the SMR concepts and designs addressed at the meeting is presented in Table 5.

Text continues on page 19.

TABLE 5. SUMMARY OF THE SMR CONCEPTS AND DESIGNS

SMR concept/design	Power level MW (thermal)	Products				Status		**Layout**			**Special features**
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) ΔK/K	Modular	
Water cooled SMRs											
SMART (ROK)	330	90	40,000	-	-	2008 – Construct 1/5th prototype	Design and construction project of 1/5th prototype plant underway	Yes	BOC: 3.0%, EOC: 1.4%	Yes	Safeguard vessel. Lifetime core option was considered.
IRIS (International consortium led by Westinghouse, USA)	1000	343 (279 desalination option)	140,000	Yes, flexible	-	2012-2015 - First-of-a-kind plant	Pre-application licensing process started in October 2002	Yes	~11% (BOL, without Xe)	Yes	Lifetime core option was considered.
CAREM (Argentina)	900	300	48,000 plus 125 MW(e) (CAREM-125)	-	-	27 MW(e) prototype in 2006 2011 – Commercial plant	Detailed design stage, licensing pre-application made	Yes	3.6% (CAREM-25 prototype core with 3.1% enrichment)	Yes	Lifetime core option was considered.
SCOR (France)	2000	630	-	-	-	2010	Conceptual design	Yes	n/a	-	Increased fuel burn-up.

SMR concept/design	Power level MW (thermal)	Products				Status		Layout			Special features
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) ΔK/K	Modular	
SAKHA-92 (Russia)	7.0	1.0	-	1.2-3.0	-	2008	Preliminary design, licensing pre-application made	Yes	n/a	-	Lifetime core (25 years). Floating NPP option was considered.
ABV-3 (Russian Federation)	16	2.5	-	5.8	-	2008	Detailed design, licensing pre-application made	Yes	n/a	Yes	Floating and land based NPP options.
ABV-6 (Russian Federation)	38	6	-	14	-	2008	Detailed design, Licensing pre-application made	Yes	n/a	Yes	Floating and land based NPP options. Lifetime core (10-12 years).
KLT-40S (Russian Federation)	2×150	60	40,000 – 100,000	40.5	-	2007	Detailed design finalized; license for construction in Severodvinsk (Russia) obtained	-	n/a	-	Floating NPP option. Lifetime core option under consideration (10-12 years).

SMR concept/design	Power level MW (thermal)	Products				Status		**Layout**			**Special features**
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) ΔK/K	Modular	
VBER-150 (Russian Federation)	350	110	Yes	Yes	-	2008	Preliminary design; no licensing pre-application	-	n/a	Yes	Floating and land based NPP options. Lifetime core (7-8 years).
RIT (Russian Federation)	161	42	80,000	Up to 110	-	2008	Preliminary design	Yes	-	-	Floating or land based nuclear desalination complex.
RUTA-70 (Russian Federation)	70	-	-	70	-	n/a	Conceptual design	-	n/a	-	Could be constructed close to the consumers.
UNITHERM (Russia)	15	1.5	-	4.65	-	n/a	Conceptual design	Yes	n/a	Yes	Lifetime core (20 years).
NIKA-70 (Russian Federation)	70	15	Yes	-	-	n/a	Conceptual design	Yes	n/a	-	A floating NPP.
ELENA (Russian Federation)	3.3	0.068	-	Up to 3	-	2010-2015	Conceptual design	Yes	Near-zero – "Self-regulating reactor"	-	Unattended operation. Lifetime core (25 years).

SMR concept/design	Power level MW (thermal)	Products				Status		**Layout**			**Special features**
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) ΔK/K	Modular	
AHWR (India)	920	300	500	-	-	2012	Basic design completed; detailed design in progress; licensing pre-application made; initiation of construction actions targeted for late 2004	-	~1%	-	Land based NPP.
PFPWR50 (Japan)	50	-	-	50	-	2012-2015	Conceptual design of the core	n/a	3% (without Xe) ~0.3% (with equilibrium Xe)	n/a	Lifetime core (10 years).
VKR-MT (Russian Federation)	890	300	-	Up to 600	-	2015-2020	Conceptual design	No	6%	No	Lifetime core (10 years).

SMR concept/design	Power level MW (thermal)	Products				Status		**Layout**			**Special features**
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) ΔK/K	Modular	
FBNR (Brazil)	30 per module	10	Yes	Yes	-	2012-2015	Conceptual design	Yes	Near zero. Reactivity margin is 'stored' in the fuel located outside the core.	Yes	Long-life core. Factory fabricated and fuelled reactor.
Gas cooled SMRs											
PBMR-400 (South Africa)	400	278	Yes	-	Yes	2010	Preliminary design	-	1.3%	X	Direct cycle gas turbine.
HTR-PM (China)	380	160	-	-	-	2010	Conceptual design	-	~1%	Yes	Indirect cycle with reheating; steam turbine
HTR (AREVA)	n/a	n/a	n/a	-	Yes	2015	Conceptual design	-	n/a	Yes	

SMR concept/design	Power level MW (thermal)	Products				Status		**Layout**			**Special features**
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) ΔK/K	Modular	
VHTR (Generation-IV)	600	n/a, optional	-	-	Yes	2030	Feasibility study	-	n/a	n/a	Advanced process heat applications.
Liquid metal cooled SMRs											
SVBR-75/100 (Russia)	280	101.5	Yes	Yes	-	2012	Detailed design stage; licensing pre-application not made yet	Yes	3.8% with UO_2 0.17% with UN	Yes	Lifetime core (6-8 years). Advanced process heat applications. Floating and land based NPP options.
SSTAR (USA)	25-50	10-25	Yes	-	-	2015	Feasibility Study	Yes	n/a	Yes	Lifetime core (target: 15-20 years)
STAR-LM (USA, Generation-IV)	400	178	Yes	-	-	2020	Conceptual design stage	Yes	~5.0%	Yes	Lifetime core (20 years). Passive load follow operation.

SMR concept/design	Power level MW (thermal)	Products				Status		**Layout**			**Special features**
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) $\Delta K/K$	Modular	
STAR-H2 (USA, Generation-IV)	400	-	8000	-	170 MW (thermal) - lower heating value	2030	Conceptual design stage	Yes	n/a	Yes	Lifetime core (20 years). Passive load follow operation.
SPINNOR	27.5/55	10/20	Yes	-	-	2020	Conceptual design stage	Yes	~0.35%	Yes	Lifetime core (25/15 years).
VSPINNOR	17.5	6.25	Yes	-	-	2020	Conceptual design stage	Yes	~0.35%	Yes	Lifetime core (35 years)
Non-Conventional SMRs											
MARS (Russian Federation)	16	Up to 6 MW	Yes	Up to 8.5	Various process heat applications	2025-2030	Conceptual design stage	Yes	3%	n/a	Lifetime core (15 or 60 years). Floating, land based or underground NPP options considered. Autonomous source of power and heat.

SMR concept/design	Power level MW (thermal)	Products				Status		Layout			Special features
		Electricity, MW (electric)	Potable water, m^3/day	District heating, MW (thermal)	H_2 production m^3/day	Target commercialisation date	Licensing status	Integral	Maximum excess reactivity in hot full power state (HFP) ΔK/K	Modular	
CHTR (India)	0.1	n/a, Brayton cycle plus thermoelectric conversion	Yes	-	Thermo-chemical processes plus electrolysis	2015	Conceptual design stage	Yes	~11%	n/a	Lifetime core (5 years). Compact power pack for remote places and for hydrogen production
CANDLE (Japan)	A fuel burn-up / fuel cycle concept for thermal and fast reactors; provides for a high degree of natural uranium utilization without the recycle and reprocessing of spent nuclear fuel. The deployment projection for CANDLE - HTGR is 2015; the deployment projection for CANDLE - fast reactors is 2040-2045. The degree of natural uranium utilization in CANDLE – fast reactors may reach 40% (absolute).										

4.2. Innovative approaches to safety

It is of special interest to SMR designers to reduce the off-site emergency planning requirements, since this could significantly enhance the economic viability and improve the public acceptance of their reactors. However, a detailed evaluation of each particular concept or project with a link to the relevant national constraints would be required to assess the effectiveness of such approach.

The insight of a presentation on the IRIS was that an off-site emergency planning necessitates the incremental infrastructure costs, such as costs of roads and bridges, and therefore comes an incentive to reduce or eliminate it. The IRIS designers intend to achieve this through a "safety by design™" approach, i.e. by the incorporation of as many inherent safety features and passive safety systems as achievable at the design stage. Such approach aims to eliminate the possibility of accidents from occurring rather than deal with their consequences, thus significantly improving the defence-in-depth and safety characteristics. Benefits may come from a simplified design or a reduced number of the required safety systems, which enables simultaneously enhancing safety and reducing the plant cost.

Similar though untitled strategies are used in the designs of SMART, CAREM, PBMR-400, AHWR, HTR-F, and VHTR (Generation-IV). The SVBR-75/100 relies to a high degree on the inherent safety features, and so do the authors of a FBNR concept.

The designs of CAREM, SMART, PBMR, SVBR-75/100, and HTR-F implement an approach where the reactor and safety systems are jointly optimized in order to ensure a cost effective safety design. The innovative methodology and tools were specially developed for this purpose.

Several innovative designs and concepts, such as PBMR, HTR-PM, PFPWR50, STAR (with ceramic structural materials), and FBNR provide for an incorporation of the refractory fuel forms or/and structural materials and promise a very large margin to fuel failure and perfect retention of fission products at high temperatures. As it was pointed out in a presentation on the PFPWR50, the promotion of this approach may help change the public attitude to nuclear energy from "NIMBY" ("Not In My Back Yard") to "CIMBY" ("Come Into My Back Yard" or "Construct In My Back Yard").

A reliable operation and, probably, a better public acceptance of SMRs could be achieved through the implementation of passive reactivity regulation and control systems, as it is done in the designs of SVBR-75/100, "STAR family", SPINNOR/VSPINNOR, CHTR, and ELENA.

Some SMR designers target to provide the total reactivity margin in a hot core so small as to secure the survival of an unprotected transient overpower with no core damage. Such approach is implemented in the designs of HTR-PM, PBMR-400, ELENA, FBNR, SVBR-75/100, SPINNOR/VSPINNOR, and VKR-MT. The acceptable reactivity margin depends on the temperature margin to fuel failure and is generally higher for the cores based on high-temperature fuel and structural materials. To reduce a reactivity margin, small lead-bismuth cooled reactors SVBR-75/100 and SPINNOR/VSPINNOR use the core configurations and material allocations optimized by design, while the high-temperature gas cooled reactors PBMR-400 and HTR-PM and a fixed bed PWR–FBNR - rely on a continuous refuelling option. Different from this, a pebble bed BWR-VKR-MT - relies mostly on the use of a high temperature fuel, which on itself secures a sufficient margin to fuel failure in many severe accidents. Very small power reactors, such as the ELENA of Russia may ensure the reactivity self-regulation throughout a very long period of unattended operation.

A certain adjustment of regulatory rules and procedures, e.g. making them technology-neutral may be necessary to realize the potential of innovative SMRs to break the economy of scale through a simplification of their design and the abandoning of some costly safety systems. For example, the authors of the "STAR family" go so far as to propose a non-nuclear safety grade balance of plant for several of their concepts. The designers of SVBR-75/100 mentioned that they target to achieve a similar goal without modifying the current regulatory rules.

To facilitate the adjustment of safety requirements and regulations, the designers of IRIS have kept informed the IAEA of their activities from an early stage and are planning to have an IAEA safety review. The need of an early involvement of regulators was noted in the presentations on the advanced high temperature gas cooled reactors HTR-PM and PBMR-400.

An experience of licensing by the prototype plant demonstration was mentioned as potentially relevant and desirable for some innovative SMR concepts. Reference was made to the experience of EBR-II sodium cooled experimental fast reactor of the Argonne National Laboratory (USA), which was in this way licensed in 1986.

4.3. Infrequent refuelling option

In a series of presentations the emphasis was on SMRs that operate without refuelling during the whole service period or require the refuelling only after a long period of operation. Such reactors were rated as capable of providing an enhanced energy security and proliferation resistance. It was mentioned that, once an SMR is factory assembled and transported to a site, the issues of safety and security in transportation become essential and, therefore, the corresponding safety rules and regulations should be in place. At the moment it may be easier to transport fuel loads for such reactors, since the regulations for a safe transportation of the reactor core or its parts are already enforced in several Member States. The transportation of an assembled reactor may be essentially equivalent to the transportation of a core load, once the latter is transported as a single unit or in a limited number of batches under strict safety and security measures, including safeguards.

A presentation on CAREM has pointed to the fact that perhaps many near-term SMRs with thermal spectrum of neutrons and traditional concept of fuel could be adjusted to a long-life core operation without on-site refuelling. This could be achieved through the reduction of their specific power and by an extensive use of burnable absorbers. However, a reduced core power density and an increased amount of burnable poisons will result in the increase of fuel costs, boosted by a higher enrichment. The situation is different in fast reactors where there is no need to increase the enrichment. The lifetime core operation is facilitated by a high conversion ratio as achieved in fast neutron spectrum (SVBR-75/100, the "STAR family") and/or by a heterogeneous core with the fertile central part securing that the importance of a newly produced plutonium is higher than of that incinerated (SPINNOR and VSPINNOR). An increase of fuel enrichment could be avoided not only in fast reactors. For example, in a thermal FBNR the reserve reactivity is stored outside the core (but inside the reactor pressure vessel) as a stock of fresh fuel particles or pebbles that are gradually moved into the core to compensate for reactivity change under burn-up. But the reduction of core power density cannot be avoided in small reactors without on-site refuelling and, therefore, innovative approaches to the design and fabrication technology are needed to secure a high degree of their economic competitiveness.

4.4. Arrangements of the overall nuclear architecture

Two presentations, one from the USA and one from the Russian Federation, addressed the complete nuclear energy systems, including both NPPs based on small reactors without on-site refuelling and regional fuel cycle centres or facilities. The insight of these presentations was that a minimization of the overall energy product cost could result from an optimization of the fuel cycle cost as well as the NPP cost, or may rely on finding an optimum system combination of the NPPs with SMRs and the associated fuel cycles.

The presentations from Russia, omit SVBR-75/100, focussed on the specific needs of low-populated North and Far East regions of the country, where the costs of electricity are essentially higher than elsewhere due to certain seasonal conditions that are severe for transportation. The autonomous NPPs with SMRs providing long-life core operation may be a preferable choice for these regions for reasons of energy supply security. One of the presentations also suggested that a decommissioning strategy should be considered when evaluating the overall cost relevant for a nuclear energy system.

A presentation from the USA addressed a scenario of global deployment of many thousands of small reactors without on-site refuelling in a system backed by the regional fuel cycle centres. This presentation linked the infrastructure changes due to an anticipated large-scale advent of nuclear power to the experience of industrial revolution of the 18th century and came up with a proposal to create a globally networked nuclear energy system providing for the flexibility in siting and applications.

The insight of a presentation from Argentina was that some developing countries for reasons of their own would insist to emplace an autonomous fuel cycle[3], so that the regional centre/long refuelling interval approach will not be right for everyone. In the discussions it was mentioned that regional infrastructure could be made compatible with the fuel cycles of those countries that master and would prefer to keep an autonomous fuel cycle option.

4.5. Fuel cycle options

Several approaches to fuel cycle services including uranium, plutonium, or thorium fuels for SMRs were presented ranging from the entire indigenous[4] fuel cycle infrastructures to services provided by the supplier and to international fuel cycle centres. The relative benefits and costs of these various approaches in terms of energy security and safeguards implementation will require an institutional as well as technical evaluation.

4.6. Application potential

All presented SMR designs provide for or do not exclude an option of offering non-electrical energy products, such as potable water, hydrogen, district heating, and others along with the electricity cogeneration. Some innovative SMR concepts, such as STAR and PBMR rely on a complex co-generation option with electricity, hydrogen and potable water being produced within a single balance of plant. Many SMR designers are aware of the fact that purposeful use of the rejected heat will improve the economic characteristics of their SMRs. In particular, high temperature SMRs could provide for an effective cost free use of the rejected heat, e.g. for potable water production. Altogether, it appears that SMRs may have a strong role to play in many non-electrical applications of nuclear energy, and therefore it was not a surprise when one of the presentations suggested that a flexible cogeneration option should be viewed as one of the major requirements to SMRs.

[3] Autonomous fuel cycle provides for an independence from the supplier, but does not assume all services to be carried out all time on a domestic basis. For example, Argentina has a capacity to produce the enriched uranium domestically, but at the moment buys it from foreign suppliers.

[4] Indigenous fuel cycle means that all fuel cycle services are provided on a domestic basis.

5. IDENTIFICATION OF COMMON TECHNOLOGY DEVELOPMENT ISSUES FOR SMRs

An enabling technology is the technology that needs to be developed and demonstrated to make a certain reactor concept viable. The enabling technology may be some key technology of a reactor core, such as certain coolant, fuel, or structural material technology; it could be a technology relevant for certain inherent safety features or passive safety design options, e.g. a core configuration to ensure the optimum set of reactivity feedbacks; likewise it might be a technology for a secondary or an auxiliary circuit, or to an overall plant configuration; finally, certain calculation technologies and data sets also fall under this definition.

The presentations at the meeting provided many examples of the enabling technologies that are common for different designs or even different lines of innovative SMRs. One obvious example is that all three near-term pressurized water SMRs, i.e., SMART, IRIS and CAREM, and the SCOR concept of the AREVA-CEA (France) rely on an integral design of primary circuit with the in-vessel location of steam generators, which is a design approach to eliminate large-break loss-of-coolant accidents (LOCA). Then, the designs of CAREM and IRIS implement the in-vessel control rod drive mechanisms (CRDM), which is to eliminate accidents with the ejection of control rods. SMART, CAREM and IRIS strongly rely on natural circulation for decay heat removal and incorporate the proven fuel design of larger capacity PWRs.

Some designs of small lead and lead-bismuth cooled reactors, e.g. SVBR-75/100 and SPINNOR target the optimum combination of reactivity feedbacks, such as a small positive or negative void reactivity effect plus a minimized burn-up reactivity swing. In particular, the presentation on SPINNOR / VSPINNOR outlined an effective enabling technology to secure such optimum combination of feedbacks: the use of a heterogeneous core with a relatively small fertile zone in its central part. This technology was shown effective for lead-bismuth reactors within the power range of 10 to 1000 MW(e).

Except for the SVBR-75/100, all lead and lead-bismuth cooled reactors require a validation of the structural materials' performance in heavy metal coolant flow and may benefit from the R&D on advanced materials for high temperature liquid metal coolant service, e.g. silicon carbide. All lead and lead-bismuth cooled SMRs provide for an option of nitride fuel, which is another common enabling technology that could be developed on a common or shared basis. Some of the designs may rely on a nitride fuel enriched by ^{15}N, and the economical penalties of such approach could also be evaluated through cooperation of the designers. Natural circulation is either a basic or a post-accidental mode of heat removal from the cores of small lead and lead-bismuth cooled reactors, and all of them envisage a passive decay heat removal to the ultimate heat sink and target the reactor capability to survive an unprotected combination of design basis accidents without core damage. All lead and lead bismuth reactor designs benefit from a high margin to coolant boiling but make special provisions to ensure the sufficient margin to coolant freezing. Some designers suggest that transportation of a lead core in frozen state may be sound from the standpoint of safety, e.g. if the transporting ship sinks, and security. All designers of lead-bismuth cooled small reactors have to deal with the ^{210}Po problem, etc.

The presentations also outlined certain enabling technologies that may be common for several different reactor lines. For example, coated particle or pebble bed type fuel is considered for use not only in high temperature gas cooled reactors, such as PBMR and HTR-PM, but also in several innovative water cooled, molten salt cooled, and even lead-bismuth cooled SMRs, such as PFPWR50 (Japan), VKR-MT (Russia), perhaps FBNR (Brazil), MARS (Russia), and CHTR (India). Some projections for the advanced high temperature structural materials of lead-bismuth cooled reactors include silicon carbide (the US “STAR family”) and pyrolithic graphite (CHTR, India). Both these materials are well qualified for the conditions of high temperature gas cooled reactors.

Another example is fixed bed core technology, in which a movable pebble-bed core is kept in the upper critical position by a coolant flow. Once the coolant flow disappears, the force of gravity moves the pebbles down to a safe sub-critical chamber. Such technology is proposed for both molten salt cooled MARS reactor of the RRC “Kurchatov Institute” (Russia) and for pressurized water FBNR of the Federal University of Rio Grande Do Sul (Brazil). The difference is that the MARS relies on a natural convection of molten salt coolant, while the FBNR provides for the use of a forced circulation. The coated particle, the pebble and the compact fuels were originally designed for the conditions of high temperature gas cooled reactors. Therefore, any case of their alternative use will require a new fuel design to be developed and demonstrated, e.g. for the compatibility of SiC-coated TRISO type fuel with water under irradiation.

Some passive decay heat removal systems, such as a water tank surrounding the reactor vessel of a lead-bismuth cooled SVBR-75/100, could be effective for many heavy metal cooled SMRs. They are also quite common to many innovative water cooled SMRs. To abandon the off-site emergency planning it may be important to develop passive heat removal systems that are effective over the whole run of a design basis accident or even an anticipated transient without scram. This may be a task important for many SMR designs representing several reactor lines.

Some technologies may be common between primary and secondary circuits of different SMRs. A remarkable example is the ‘flibe’ secondary circuit of the STAR-H2 reactor, which is a high temperature molten salt loop that transfers heat from the lead-based primary circuit to the multi-application balance of plant. A molten salt technology for such loop may have common points with the primary coolant technology for a molten salt cooled MARS reactor of Russia.

Table 6 presents a crosscut of the enabling technologies for SMRs addressed at the meeting. This table points to some technology development areas that could be of common interest to several SMRs.

TABLE 6. CROSS-CUT OF THE ENABLING TECHNOLOGIES FOR SMRs

	Enabling technology	SMR concepts and designs of relevance
1	Integral design of primary circuit	Near-term PWRs: SMART, IRIS, CAREM, SCOR, SAKHA-92, ABV, RIT, FBNR Lead or lead-bismuth cooled reactors: SVBR-75/100, “STAR family”, SPINNOR/VSPINNOR Non-conventional designs: MARS (RRC “Kurchatov Institute”), CHTR, ELENA
2	In-vessel control rod drive mechanism	Near-term PWRs: IRIS, CAREM, SCOR

	Enabling technology	SMR concepts and designs of relevance
3	Natural circulation in primary circuit	Near-term PWRs: CAREM-25, SMART, SAKHA-92, ABV, UNITHERM, IRIS-50 (a reduced power version of IRIS) Advanced heavy water reactor: AHWR Lead or lead-bismuth cooled reactors: “STAR family”, VSPINNOR Non-conventional designs: CHTR, ELENA, Pebble-bed fuel molten salt coolant fixed bed reactor MARS
4	Materials for high temperature liquid metal coolant service	Lead or lead-bismuth cooled reactors: “STAR family”, SPINNOR/VSPINNOR; further evolution of the SVBR-75/100 towards higher primary circuit outlet temperatures Non-conventional designs: CHTR Innovative fuel cycle concepts: CANDLE for lead-bismuth cooled SMRs
5	Technologies to ensure lifetime core operation	Small PWRs: RUTA-70, SAKHA-92, ABV, KLT-40S with long-life core, VBER, RIT, UNITHERM, NIKA-70, FBNR, PFPWR50 Options of long-life core for near-term PWRs: IRIS, CAREM, SMART Small BWRs: VKR-MT Lead or lead-bismuth cooled small reactors: “STAR family”, SVBR-75/100, SPINNOR/VSPINNOR Non-conventional designs: MARS, CHTR, ELENA
6	Coated particle fuel technology	High temperature gas cooled reactors: PBMR-400, HTR-PM, HTR-F Pebble bed BWR: VKR-MT, a bed of coated particles cooled by a lateral flow of the coolant in fuel assemblies with perforated collector walls Lead-bismuth cooled reactors: the “STAR family” – materials tried for

	Enabling technology	SMR concepts and designs of relevance
		high temperature liquid metal coolant service include SiC, a material common for a load-bearing coating layer in TRISO fuel
7	Pebble bed or prismatic block HTGR type fuel technology, including coated particles	High temperature gas cooled reactors: PBMR-400, HTR-PM Small PWRs: FBNR – fixed bed modular PWR; one of fuel options considered is a pebble bed fuel PFPWR50 – pellets of coated particles in carbon matrix inside the conventional PWR type cladding tubes Non-conventional designs: MARS – fixed-bed fuel molten salt coolant reactor CHTR – compact high temperature reactor: prismatic block HTGR type fuel and lead bismuth coolant CANDLE in application to high temperature gas cooled reactors with prismatic block fuel
8	Refuelling by pebble transport	High temperature gas cooled reactors: PBMR-400, HTR-PM Small BWR: VKR-MT, a bed of coated particles reloaded in a single batch by hydraulic transport Small PWR: FBNR – one of the options is that fuel elements are loaded to the modules by a pebble transport system
9	Fixed bed core technology	Small PWR: FBNR – fixed bed modular PWR with forced circulation of water coolant Non-conventional designs: MARS – fixed-bed fuel reactor with natural circulation of molten salt coolant
10	Molten salt coolant technology	Lead bismuth cooled small reactor: STAR-H2 - the ‘flibe’ (molten salt) secondary loop Non-conventional designs: MARS – fixed-bed fuel molten salt

	Enabling technology	SMR concepts and designs of relevance
		coolant reactor
11	Co-generation with multiple non-electrical applications	High temperature gas cooled reactors: PBMR-400, HTR-F Lead bismuth cooled small reactors: “STAR family”
12	The technology of barge-mounted NPPs	Small PWRs: ABV, KLT-40S, VBER, RIT, NIKA-70, an option for IRIS Lead-bismuth cooled reactors: an option for SVBR-75/100 Non-conventional designs: a floating NPP option is envisaged for MARS
13	Design approaches and materials for long-life cores	All SMRs with core lifetime exceeding 7-8 years, see Table 5

6. IDENTIFICATION OF COMMON INFRASTRUCTURE ISSUES

Apart from design and technology development issues, the presentations at the meeting suggested several important trends of infrastructure furthering to support the deployment of innovative SMRs. Certain infrastructure changes, such as establishment of design certification/licensing reciprocity regimes between different countries, creation of legal and institutional provisions for fuel leasing, early involvement of regulators to secure that rules and procedures matching an innovative safety design are ready on time, harmonization of the industrial standards/codes and the regulatory rules/procedures could be of benefit for all innovative reactors, not SMRs only. However, some infrastructure changes were mentioned as being of special benefit namely to SMRs, among them:

- Reestablishment of the rules and practice of licensing by the prototype demonstration;
- Establishment of legal provisions and the insurance scheme for a transit of fuel loads or factory fabricated SMRs through the territory of a third country;
- Provision of international guarantees of sovereignty for countries that would prefer to lease fuel.

Regarding the last item, a study on how the core lifetime in operation without on-site refuelling could affect the issue of sovereignty was recommended. Finally, the two presentations from the USA and the Russian Federation made a focus on the desirable infrastructure changes associated with the creation and operation of multinational fuel cycles. It was outlined that international fuel cycles could, probably, be created on a regional or interregional basis, perhaps starting from international repositories of waste. At the same time, it was mentioned that some countries that already master an autonomous nuclear fuel cycle have very cautious attitude to the idea of multinational fuel cycles as they envisage the associated threats to their sovereignty and energy security, as well as economic disadvantages that may stem from resulting insufficient competition.

Table 7 summarizes the suggestions on infrastructure development made at the meeting.

TABLE 7. CROSS-CUT OF INFRASTRUCTURE DEVELOPMENT NEEDS

	Desirable infrastructure change	Relevant SMR concepts and designs
1	Early involvement of regulators in the design development process	Near term PWRs: IRIS, SMART High temperature gas cooled reactors: HTGR, HTR PM
2	Development of technology-neutral safety requirements (or dedicated safety requirements for each specific reactor line)	High temperature gas cooled reactors: all designs Lead bismuth cooled SMRs: all designs Non-conventional SMRs: all designs
3	Reestablishment of rules and practice of licensing by the prototype demonstration	All designs except near-term PWRs
4	Establishment of legal and institutional provisions for fuel or NPP leasing	Small reactors without on-site refuelling: "STAR family", ELENA, RUTA-70, MARS, VKR-MT, SAKHA-92, ABV, KLT-40S with long life core, VBER, RIT, SVBR-75/100, UNITHERM, NIKA-70, FBNR, SPINNOR/VSPINNOR, CHTR, long life core design options for IRIS and CAREM
5	Provision of liability and insurance arrangements for the transit of fresh and spent fuel loads and/or factory fabricated and fuelled reactors through the territory of a third country	All designs that fit into the category of small reactors without on-site refuelling
6	Provision of the internationally secured guarantees of sovereignty to those countries that would prefer to lease fuel for their NPPs	Many SMRs*
7	Development of legal and institutional provisions for multinational fuel cycles	Small reactors without on-site refuelling*

* The designers of small reactors without on-site refuelling argue that an infrequent refuelling interval is on itself a certain guarantee of sovereignty for those countries who would prefer to lease fuel.

7. REVIEW OF PASSIVE SAFETY DESIGN OPTIONS FOR SMRs

The presentations and discussions at the meeting defined the following focus areas relevant for passive safety design options and inherent safety features of innovative SMRs:

1. Inherent safety features of reactor core, including:

- Combinations of reactivity effects;
- Features contributing to high degree of design robustness, e.g. offered by advanced fuel designs providing enhanced retention of fission products and/or large margin between operating and damage states;
- Features offered by advanced coolants and structural materials;

2. Passive safety design options:

- Passive cooling systems based on natural convection;
- Passive reactivity control systems, including passive shut down systems;
- Passive decay heat removal systems, including those being efficient during the entire duration of a design basis accident, including an anticipated transient without scram;
- Passive systems for mitigation of severe accident consequences.

The following issues were identified as important and requiring further R&D:

- Combined action of active and passive systems;
- Increasing the defence-in-depth by implementing a ‘safety by design’ approach;
- Passive safety design options and cost benefit design optimization.

Table 8 gives a crosscut of certain groups of the inherent safety features and passive safety systems for SMRs addressed at the meeting. This table points to some approaches in passive safety design implementation that may be common for a number SMRs.

Many designers identified an intention to license their innovative SMRs with the reduced or eliminated off-site emergency planning requirements (IRIS, SMART, CAREM, SAKHA-92, ABV, KLT-40S with long-life core, VBER, RIT, FBNR, PBMR-400, HTR-PM, AHWR, SVBR-75/100, and the “STAR family”). However, it was confirmed that no example of such licensing exists at the moment.

TABLE 8. CROSS-CUT OF PASSIVE SAFETY DESIGN OPRIONS FOR SMRs

	Inherent safety features or passive safety systems that provide:	Relevant SMR designs
1	Minimum reactivity margin in the core	Lead or lead-bismuth cooled small reactors: SVBR-75/100, SPINNOR/VSPINNOR – minimum reactivity margins are provided through design optimization CANDLE burn-up strategy for HTGR and lead-bismuth cooled SMRs
2	Perfect confinement of fission products at High temperatures and a high margin to fuel failure	High temperature gas cooled reactors: PBMR-400, HTR-PM Small BWR: VKR-MT - a bed of coated particles cooled by lateral flow of the coolant in fuel assemblies with perforated collector walls Small PWRs: FBNR – fixed bed modular PWR; PFPWR50 – pellets of coated particles in pyrolized graphite matrix within conventional PWR type claddings Lead-bismuth cooled small reactors: SVBR-75/100, “STAR family” with new, high temperature structural materials, e.g. SiC Non-conventional designs: MARS – a fixed-bed fuel molten salt coolant reactor; CHTR – a prismatic block fuel lead bismuth coolant reactor
3	Passive reactivity control and reactor shut down	Lead-bismuth cooled small reactors: SVBR-75/100, “STAR family”, SPINNOR/VSPINNOR Non-conventional designs: CHTR, ELENA
4	Passive decay heat removal	All SMRs considered at the meeting
5	Mitigation of severe accident consequences	Near-term PWRs: IRIS, SMART, CAREM, SAKHA-92, ABV, KLT-40S with long-life core, VBER, RIT Small PWR: FBNR – fixed bed modular

	Inherent safety features or passive safety systems that provide:	Relevant SMR designs
		PWR Advanced heavy water reactor: AHWR Lead-bismuth cooled small reactors: SVBR-75/100, “STAR family” High temperature gas cooled reactors: HTR-F
6	Minimization of safety-related costs through design optimization	Near-term PWRs: SMART, CAREM, IRIS, SAKHA-92, ABV, RIT Small PWR: FBNR – fixed bed modular PWR Lead-bismuth cooled modular reactor: SVBR-75/100 Non-conventional designs: CHTR

8. DEFINITION OF SMALL REACTORS WITHOUT ON-SITE REFUELLING

The participants of the meeting confirmed that a previously used term ‘Small Reactors[5] without On-site Fuelling’ should be changed to ‘Small Reactors without On-site Refuelling’. They also agreed that a “refuelling” could be defined as “the removal and/or replacement of either fresh or spent, single or multiple, bare or inadequately confined nuclear fuel cluster(s) or fuel element(s) contained in the core of a nuclear reactor”. With this, the infrequent replacement of well-contained fuel cassette(s) in a manner that prohibits clandestine diversion of nuclear fuel material could be exempted.

The design goals for small reactors without on-site refuelling were defined as follows:

1. Small reactor without on-site refuelling should have the following essential features:

- Capability to operate without refuelling for a reasonably long period consistent with the plant economics and energy security;
- Minimum inventory of fresh and spent fuel being stored at the site outside the reactor during its service life;
- Enhanced level of safety, consistent with the scale of global deployment of such reactors, through wider implementation of inherent and passive safety features and systems;
- Economic competitiveness for anticipated market conditions and applications;

[5] According to the classification adopted by the IAEA, ‘small reactors’ are reactors with the equivalent electric power less than 300 MW.

- Difficult unauthorized access to fuel during the whole period of its presence at the site and during transportation, and design provisions to facilitate the implementation of safeguards;
- The capability to achieve higher manufacturing quality through factory mass production, design standardization and common basis for design certification.

2. Small reactor without on-site refuelling may have the following additional desirable features:

- Factory fabrication and fuelling to facilitate delivery of a sealed core to the plant site;
- Capability to survive all postulated accident scenarios, including those caused by natural or human-induced external events, without requiring emergency response actions arising out of unacceptable radiological consequences in the public domain and without compromising the transportability of reactor back to the manufacturers;
- An overall reactor and fuel cycle enterprise that is highly unattractive for weapons purposes, e.g. offering limited overall amount of material, high degree of contamination providing noticeable radiation barriers, incorporating fuel forms that are difficult to reprocess and/ or types of fuel that make it difficult to extract weapons-grade fissile material;
- A variety of applications, including generation or co-generation of electricity, production of heat, potable water, or hydrogen;
- A variety of options for siting, including those close to population centres, as well as remote and hardly accessible areas, dispersed islands, etc;
- Simplified operation procedures and robustness with respect to human errors;
- Minimum reliance on sophisticated local infrastructure;
- An overall reactor and fuel cycle enterprise that contributes to effective use of resources in a sustainable way.

9. CONCLUSION

The IAEA Technical Meeting on Innovative Small and Medium Sized Reactors: Design Features, Safety Approaches and R&D Trends demonstrated a continued interest of Member States to the design and technology development for such reactors. The major conclusions of the meeting are as follows:

(1) The majority of innovative SMR designs are light water cooled, and the following major groups of such reactors were addressed at the meeting:

- Integral type PWRs targeted at near term deployment. All reactors in this group provide design solutions to exclude the possibility of certain accidents, e.g. large-break LOCA or control rod ejection. They also incorporate cost-benefit design optimization to beat the economy of scale;
- Modular integral or loop type PWRs for barge-mounted NPPs. These reactors, coming mostly from the Russian Federation, are factory fabricated and fuelled, and provide for design standardization and mass production to increase their competitiveness. They

make a full use of the multi-year operation experience of the reactors of nuclear icebreakers and submarines, and incorporate many inherent safety features and passive safety systems. One design of a barge-mounted NPP, the KLT-40S of Russia, was completed and a license for plant construction in Severodvinsk has been obtained;

- Small battery-type reactors for heat and electricity supply to isolated settlements in remote areas. These designs from Russia provide for a very small core power density, which makes it possible to rely on reactivity self-regulation and passive shut down during a very long period of unattended operation;
- An advanced heavy water reactor (AHWR) from India, which is the evolution of a CANDU type pressurized heavy water reactor towards light water cooling, combined use of the uranium, plutonium and thorium fuel, and strong reliance on the inherent and passive safety design options. This reactor is designed as part of an overall nuclear energy system based on the closed nuclear fuel cycle and also including fast reactors and accelerator driven systems for fuel breeding and waste transmutation;
- Several concepts of water cooled SMRs with coated particle or pebble bed HTGR type fuel. Being capable to confine fission products perfectly at very high temperatures, such fuels are targeted to avoid heat exchange crisis, to reduce heat energy stored in the core, and to eliminate significant radioactivity releases in beyond design basis accidents. The suggested modes of fuel use vary from coated particles in a graphite matrix within conventional zirconium alloy claddings to a pebble bed of coated particles directly cooled by lateral coolant flow, to a movable bed of spherical fuel elements (the fixed bed concept). Furthering of these concepts will require certain R&D on fuel design and qualification for the conditions of water cooled reactors;

All designers of the innovative water cooled SMRs would attempt to reduce or eliminate the off-site emergency planning requirements in licensing;

(2) The designs of innovative high temperature gas cooled reactors are firmly rooted in the past experience of the HTGR reactors. Incorporating certain limits on core dimensions and core power density, such SMRs have a large temperature margin and an exceptional passive heat removal capability, which allows them to survive many unprotected design basis accidents and secures minimum radioactivity releases in severe accidents. A sound safety design of the HTGRs makes their designers argue whether such traditional attributes of LWR plants as the reinforced pressure containment and the two independent and diverse reactivity control systems are needed. A simplification or abandoning of some safety and safety-related structures and systems could reduce the NPP cost, which is an urgent issue for high temperature gas cooled reactors. Like water cooled SMRs, such reactors are candidates for near-term deployment, and their designers also attempt to reduce or eliminate an off-site emergency planning;

(3) Being capable to operate in a self-sustainable regime on nuclear fuel or as breeders, the liquid metal cooled SMRs are usually associated with further stages of nuclear power, when the deficiency of natural fissile isotopes may facilitate decisions on closed nuclear fuel cycles. The meeting addressed only lead and lead bismuth cooled innovative SMRs, and the summary of the design approaches is as follows:

- All designs in this group incorporate reactivity self-regulation achieved through an optimum combination of reactivity effects provided by the design. A common approach is to ensure minimum reactivity margins in hot core by minimizing the burn-up reactivity swing and ensuring that the void reactivity effect is below a single effective delayed neutron fraction or is eliminated by design. The safety analyses prove that such reactors may survive an unprotected transient overpower combined with one or two other designs basis accidents;
- Some lead and lead-bismuth cooled SMR designs, e.g. the STAR systems from the USA target the use of advanced high temperature structural materials, such as silicon carbide. If the reliability of such materials is qualified, they would ensure a higher temperature margin to fuel failure in accidents other than transient overpower. Moreover, the use of such advanced materials could broaden the application potential of lead or lead-bismuth cooled SMRs to include hydrogen production and other advanced process heat applications;
- For reasons mentioned above, all designers of the innovative lead and lead-bismuth cooled SMRs target not only the reduced or eliminated off-site emergency planning but also a non-nuclear safety grade balance of plant. The latter may help reduce capital costs and would also increase local participation when an SMR is deployed in a developing country;
- A different deployment strategy was outlined for the SVBR-75/100 of the Russian Federation. In its design and technical features the first-of-a-kind SVBR-75/100 will not depart from the solutions proven by the operation experience of its submarine prototypes. In particular, the design would provide for a lifetime core operation of 7–8 years in a conventional uranium fuel cycle with the initial fuel enrichment below 20%. Such strategy could make SVBR-75/100 suitable for a near-term deployment, particularly for the renovation of the Russian VVER-440 units withdrawn from operation. Furthering of the SVBR-75/100, e.g. to extend its core lifetime or to adjust it to a fuel cycle option dominating at the time, will be performed through subsequent design modifications;

(4) The meeting also addressed several concepts of non-conventional SMRs, e.g. the Russian MARS concept of a fixed bed molten salt cooled small reactor with air turbine cycle in the secondary circuit. The remarkable example of a non-conventional approach to SMR design was provided in a presentation on the compact high temperature rReactor (CHTR) from India. This small battery-type reactor for power supply in remote areas or for hydrogen production combines several technologies originating from different reactor lines: lead-bismuth coolant, prismatic HTGR type fuel, beryllium oxide moderator, and ^{233}U-Th fuel. An intention to combine the technologies from different reactor lines was also observed in several concepts of light water reactors with HTGR type fuel and in those concepts of lead bismuth cooled SMRs that target the use of advanced ceramic structural materials. Such tendencies point to the need of a more close cooperation between the designers of SMRs belonging to different reactor lines;

(5) Out of 30 innovative SMRs addressed at the meeting, more than 18 designs encompassing all reactor lines are small reactors without on-site refuelling. These are the reactors that could operate without reloading and shuffling of fuel for a reasonably long period, from 5 to 60 years for the designs considered at the meeting. Such mode of operation could simplify the implementation of safeguards, minimize the adverse environmental impacts, and provide

certain guarantees of sovereignty to those countries that would prefer to lease fuel. The discussions at the meeting produced the following vision of a potential role of such reactors:

Small reactors without on-site refuelling when combined with an appropriate fuel cycle infrastructure may offer a solution for implementation of adequate safeguards in scenario of large-scale global deployment of nuclear energy, as will be essential for providing long-term energy security to many nations in an environmentally benign way;

(6) All presented SMR designs provide for or do not exclude a flexible offer of non-electrical energy products, such as potable water, hydrogen, district heating, and others along with the cogeneration of electricity;

(7) Some designers of small reactors without on-site refuelling presented their vision of future nuclear energy systems with such reactors and centralized, e.g. regional fuel cycle centres, perhaps under an international control. The designers of other SMRs advocated national approaches to fuel supply, such as indigenous or autonomous fuel cycles. It was also mentioned that different approaches to nuclear fuel cycle may be made complementary and could coexist on a competitive basis;

(8) An innovative approach to fuel burn-up management, called CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor), was presented. This approach is to organize nuclear fuel burn-up in way similar to a conventional candle-burn. Periodical axial shuffling of nuclear fuel is needed to realize the CANDLE in a reactor. Burnable poisons are used to control the burn-up front in thermal reactors, while in fast reactors such control is achieved through breeding. The maximum benefit from CANDLE implementation is in fast reactors where, on a theoretical basis, up to 40% of natural uranium could be burned out without the recycle and reprocessing of spent nuclear fuel. However, for technological reasons it is very difficult to realize CANDLE in fast reactors. The CANDLE is also applicable to high temperature gas cooled reactors, where it could be implemented with fewer difficulties but with fewer benefits too.

One of the objectives of the meeting was to support the preparation of an IAEA report on the status of innovative SMR designs and other IAEA activities for SMRs. This purpose produced a number of certain recommendations that are summarized in short below:

(a) With a variety of reactor designs developed worldwide, it was recommended that a new status report on the innovative SMR designs and a report on small reactors without on-site refuelling identify the enabling technologies that are common to different designs and lines of such reactors, in order to encourage and facilitate broader cooperation between their designers;

(b) For the review of passive safety design options it was recommended to make a focus on the following topics:

- Combined action of active and passive systems;
- Passive safety design options and cost benefit design optimization;
- Increasing the defence-in-depth by implementing a ‘safety by design’ approach.

The last topic was found strongly related to an option of NPP licensing with reduced or eliminated off-site emergency planning requirements. It was recommended to re-examine such option for the innovative SMRs developed currently by considering both the required institutional changes and the accident sequences that need to be eliminated to achieve this objective;

(c) For small reactors without on-site refuelling, it was recommended that “refuelling” is defined as “the removal and/or replacement of either fresh or spent, single or multiple, bare or inadequately confined nuclear fuel cluster(s) or fuel element(s) contained in the core of a nuclear reactor”. It was also suggested that an infrequent replacement of well-contained fuel cassette(s) in a manner that prohibits clandestine diversion of nuclear fuel material is exempted from this definition.

(d) Also for small reactors without on-site refuelling, a study on how the core lifetime could affect the issue of sovereignty for a country that would prefer to lease fuel was recommended.

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ANNEXES
1–13

ANNEX 1

MAIN RESEARCH AND DEVELOPMENT ACTIVITIES FOR SMRs IN ARGENTINA

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Abstract. Major Argentinean Project in the area of Small and Medium Sized Reactors (SMRs) is CAREM. CAREM project involves technological and engineering solutions, as well as several innovative design features that should be properly demonstrated during the design phase. Also, specific codes used for modelling of systems related to safety in the process of design optimisation should be verified and validated with the use of benchmark and/or experimental data acknowledged worldwide, in order to build confidence in their results. This paper describes principal issues of an R&D programme ongoing as part of the design phase of CAREM project and including the design and construction of several experimental facilities and engineering mock-ups. Another important project is CARA, which concentrates on the design and development of an advanced fuel assembly for heavy water reactor (HWR). Major development activities of CARA are described in brief also.

1. CAREM PROJECT

The Argentinean CAREM project [1] provides for the development, design and construction of an advanced, simple and small Nuclear Power Plant (NPP).

The idea of design cycles, developed to ensure that final product is capable of meeting specific requirements, has been applied in different frameworks that involve several steps from the conceptual design to various products, such as systems, equipment, design codes or technological processes. From the early stages of CAREM project, engineering is being conducted according to a "globally planned" sequence of this design cycle, in which two general stages may be recognized:

(a) Conceptual / basic design, and experimental activities to support this design;
(b) Detailed design, and experimental activities for validation/qualification of components.

The activities are carried out in order to construct and operate CAREM-25 plant, which will serve as a demonstration prototype for the CAREM concept.

Within CAREM project, the effort has been focused mainly on the nuclear island, i.e. internals of the containment and safety systems, where several innovative design solutions require R&D within the first stage, in order to assure that they comply with functional requirements. These are mainly the solutions for Reactor Core Cooling System (RCCS), Reactor Core and Fuel Assembly, Reactor Pressure Vessel Internals (RPVI), and First Shutdown System (FSS). To fulfil project requirements, an extensive experimental plan has been prepared that includes the design and construction of several experimental facilities.

An effort is planned for the systems/devices that require RD&D only within the second stage of a design cycle (qualification, or adaptation of a proven solution). Such systems/devices may actually be not innovative in their features, but anyway require certain development effort to meet the overall requirements of project engineering.

The RCCS modelling and qualification are supported by the tests performed in a High Pressure Natural Circulation Rig (CAPCN) and covering thermal hydraulics and reactor control and operating techniques. The CAPCN rig reproduces all dynamic phenomena of the RCCS, except for three-dimensional effects.

The Core Design involves various aspects, e.g. study of thermal margins, neutronic modelling, structural, mechanical and fuel assembly design. Neutronic modelling needs may be covered by benchmark data available worldwide and by the data from the RA-8 Critical Facility. For Fuel Element Design, the Atomic Energy Commission of Argentina (CNEA) has vast experience in nuclear fuel technology, and structural and hydrodynamic tests are being carried out at low and high pressures rigs.

Mock-up facilities are being constructed to support mechanical design of the core (structural, dynamic, seismic, etc.) and other RPVI. They represent sections of the core and include one vertical full-scale model with the supporting barrel and its Kinematics Chain.

The FSS or, more specifically, Control Rod Drives (CRD) offers a good example of the design process for an innovative device that comprises both stages of the design cycle. The experimental programme is underway for both design and qualification stages.

Brief description of some of the more important tasks for the current and future stages of CAREM project and of relevant experimental facilities is provided below.

2. DYNAMIC TEST OF RCCS

The purpose of the Natural convection high-pressure loop CAPCN is mainly to study the thermal-hydraulic dynamic response of CAREM primary loop, including all coupled phenomena that may be described by one-dimensional models. This includes the validation of calculation codes on models of the rig, and the extension of validated models to the analysis of the CAREM reactor. Activities were performed and are on going in order to validate thermal-hydraulic tools.

The CAPCN rig (Fig. 1) resembles the primary loop (with self-pressurized natural circulation) and the helical once-through steam generator of CAREM reactor, while the secondary loop is designed only to produce adequate boundary conditions. Operational parameters are reproduced for intensive magnitudes (pressure, temperature, void fraction, heat flux, etc.) and scaled for extensive magnitudes (flow, heating power, cross-sections, etc.). The rig height was kept approximately at a 1:1 scale.

The heating power may be regulated up to 300 kW by the operator or by a feedback loop of primary pressure. The secondary loop pressure and cold leg temperatures are controlled through valves. The pump regulates the flow. The condenser is of air-cooled type with airflow control. Control of the actuators (heaters, valves, pumps, etc.), data acquisition and operating follow up are carried out from a control room through a PC based, multi-node software (flexible enough to define any feedback loop).

Most of the tests [2] consist of an initial self-steady state in which a pulse-wise perturbation induces a transient. In this case the perturbation is a thermal unbalance as severe as possible, e.g. thermal power increase by 12 kW (about 5% of the full power) during 150 seconds. Primary pressure and circulating flow evolve mildly, with increases below 2 and 3% respectively, and primary temperatures hardly notice the perturbation. Therefore steam generation remains quite stable during the whole transient, a remarkable feature of the Steam Supply System (Fig. 2).

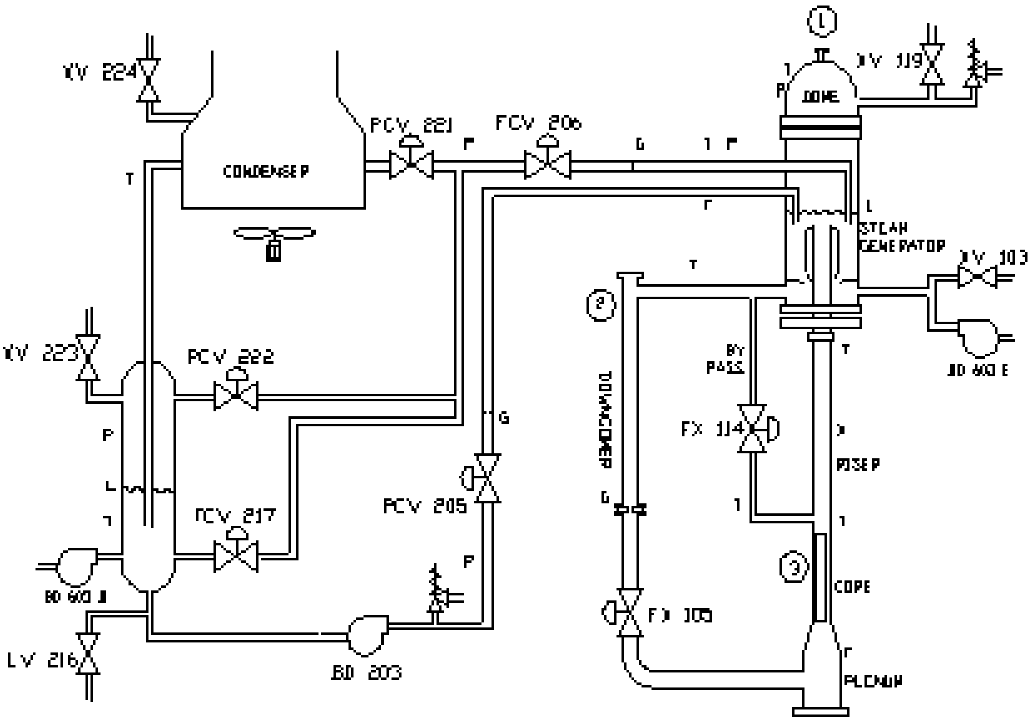

Fig. 1. Simplified process and instrumentation diagram of CAPCN.

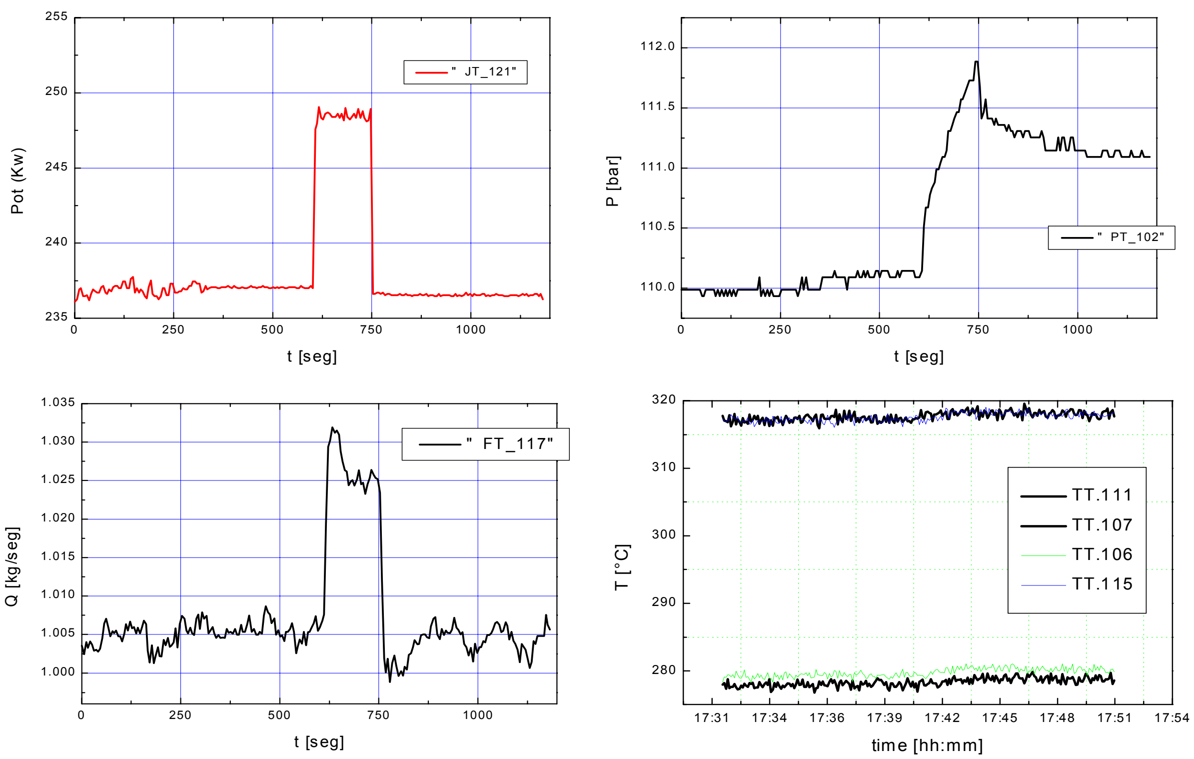

FIG. 2. Transient 1: Plus 12 kW perturbation during 150 seconds.

3. CRITICAL HEAT FLUX TESTS AND THERMAL MARGINS

The thermal hydraulic design of CAREM reactor core was carried out using an improved version of the 3D two fluid model realized in the THERMIT code. In order to take into account strong coupling of the thermal-hydraulic and neutronic characteristics of the core, THERMIT was linked with the neutronic code CITVAP. This coupled model makes it possible to produce a 3D map of power and thermal-hydraulic parameters at any moment of the burn-up cycle.

Prediction of the thermal margins of fuel elements to a harmful phenomenon like critical heat flux (CHF) during normal operation and transients is considered to be of the utmost importance. Mass flow rate in the core of the CAREM reactor is rather low compared to typical light water reactors and therefore available correlations or experimental data are not completely reliable in the range of interest. Thus, analytical data should be verified by ad-hoc experiments.

The experiments were conducted in thermal-hydraulic laboratories of the Institute of Physics and Power Engineering (IPPE, Obninsk, Russian Federation). The experimental program [3] was targeted at the generation of a substantial database and at the development of a prediction methodology for CHF applicable to the CAREM core and covering a wide range of thermal hydraulic parameters around the point of normal CAREM-25 operation. Most of the tests were performed using a low-pressure Freon rig, and their results were later extrapolated to water conditions through scaling models. Finally, a reduced set of tests was performed in water at high pressure and temperature to validate the method for scaling.

Different test sections were assembled to simulate different regions of the fuel element as well as radial uniform and non-uniform power distributions. A bundle of 35% of the full length was tested to obtain CHF data under average sub-cooled conditions. More than 250 experimental points under different conditions were obtained in the Freon loop, and more than 25 points were obtained in the water loop.

4. FUEL ASSEMBLIES

The R&D tasks on this subject cover mainly the following two issues:

- Improvement and extension of the simulation models of BACO code, which may be categorized under stage one of the design cycle;

- Verification, evaluation and qualification of the designs, which falls under stage two of the design cycle.

The BACO code [4] (BArras COmbustibles, "Fuel Rods" in Spanish) performs a best-estimate computer simulation of the principal thermal-mechanical phenomena that occur within a nuclear fuel rod during burn-up process. It simulates fission product generation and migration, fission gas release, in-cladding pressure build-up, pellet deformation, crystallographic grain growth, stress-strain state, pellet-cladding interaction, etc.

This code has already been developed and verified on the data of PHWR fuel assemblies produced in Argentina. In order to cover enriched uranium fuel assemblies some new models had to be introduced and others had to be modified. These included the influence of high burn-up on thermal conductivity of UO_2, thermal conductivity in the pellet-cladding gap (influence of Xenon at high burn-up), and the migration of porosity (densification and restructuring).

These new models were validated through the participation in a Co-ordinated Research Project (CRP) of the International Atomic Energy Agency [5]. This CRP, called FUMEX Program, facilitates the validation by sharing the experimental information on operating conditions and requirements to a certain fuel and by comparing "blind simulation" results with experimental measurements.

BACO code combined with the International Fuel Performance Experiments Database (of the OECD Nuclear Energy Agency) should cover all validation and evaluation requirements to fuel rod design.

The fuel assemblies (Fig. 3) and absorbing clusters are being subject to a series of qualification tests, including standard mechanical evaluations and hydraulic tests. The latter comprise:

- Tests in a Low Pressure Rig to evaluate pressure-losses, flow-induced vibrations and general behaviour of fuel assembly;
- Endurance tests in a High Pressure Loop to examine wear-out and fretting issues.

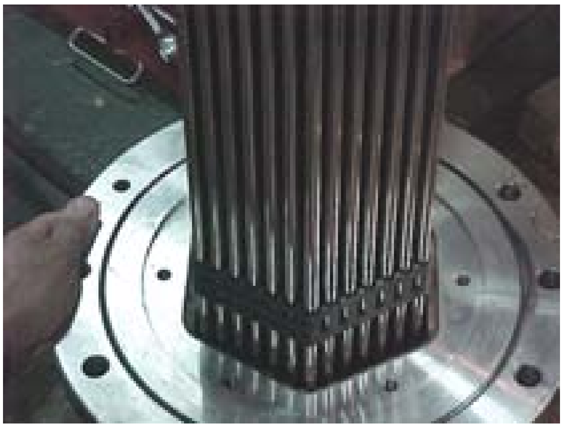

FIG. 3. Fuel assembly entering the Low Pressure Rig for pressure-loss evaluations.

5. NEUTRONIC TESTS AND BENCHMARKING

The RA-8 critical facility has been designed and constructed to measure neutronic parameters typical of the CAREM core. It provides a reactor shielding block and a reactor tank that can be adapted to contain custom designed reactor cores. Experiments were performed using fuel rods of the same radial geometry and pitch as in the CAREM-25 fuel element. Components of the neutronic calculation lines were validated with the use of data for VVER type reactors obtained in the experiments at ZR-6 Research Reactor (Central Research Institute for Physics, Academy of Sciences, Hungary) and data for PWR critical experiments.

6. HYDRAULIC CDR TESTS

One of the more innovative systems within the CAREM concept is in-vessel Hydraulic Control Rod Drive (HCRD). Two designs are under development for HCRD: "Fast

Extinction" and "Adjust & Control" CRDs, with the latter one posing major design related challenges (Fig. 4). The designs embrace mechanical and thermal-hydraulic innovations, so that feasibility of the concept should be demonstrated before it could to be included in the reactor engineering.

On the other hand, their operational functions of Adjustment and Control and of Fast Extinction are parts of one of the most important safety systems of the reactor: the First Shutdown System (FSS). Therefore a complete experimental program including both "experiment-aided design" and qualification tests is necessary to secure high reliability performance together with low maintenance requirements.

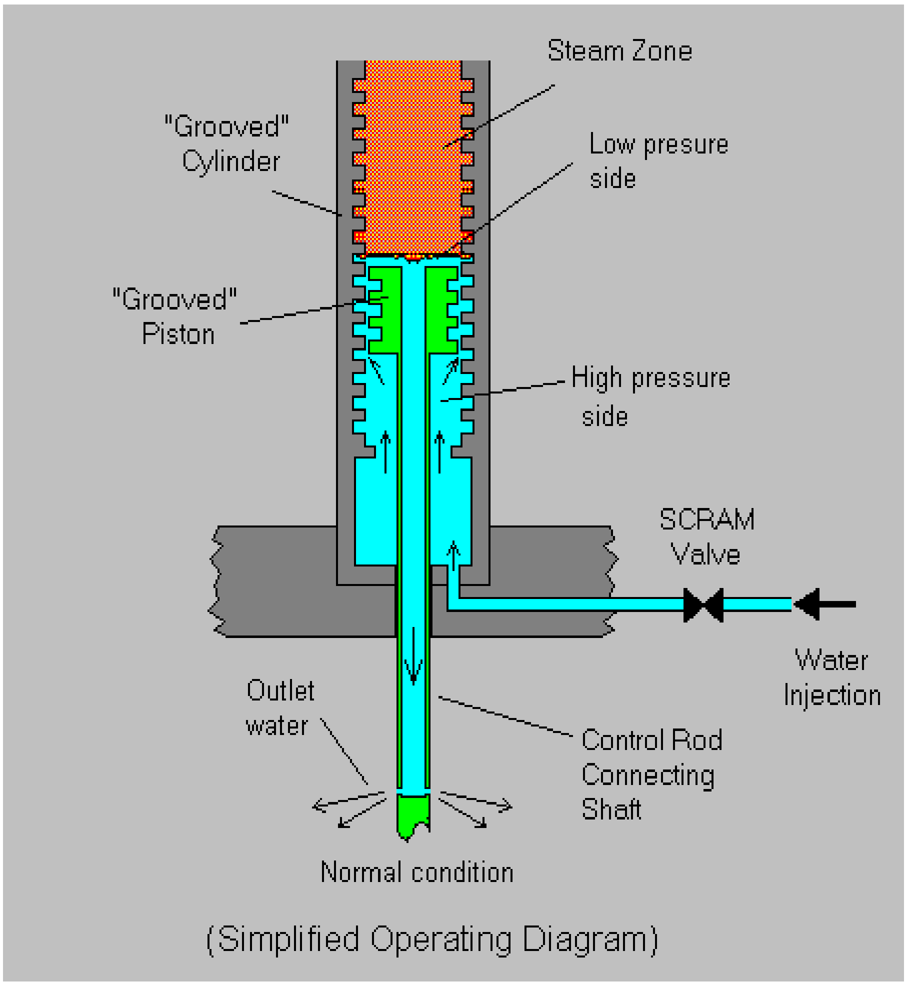

FIG. 4. HCRD Adjust and control system.

The development plan refers to four distinct stages and includes the construction of several experimental facilities to reach testing of the system performance under actual RPV operating conditions. The four different stages and their facilities are:

Preliminary tests (conceptual verification)

The aim of these tests was to prove the feasibility of a theoretically suggested approach, to get a first shot of some of the most sensitive controlling parameters, and to determine spot points to be focused upon during design. Tests were undertaken on a rough device and produced promising results that are in good agreement with the preliminary modelling data.

First prototype tests

This stage was conducted to determine preliminary operating parameters of a full-scale mechanism, as a first approach towards detailed engineering. The parameters included: range of flow, modes of producing hydraulic pulses, etc. Manufacturing hints to simplify the first design and reduce its costs were also defined and implemented. Tests were carried out in a craftily built rig and as an outcome it was decided to attribute the regulating and fast-drop functions to different devices.

Low pressure CRD rig

This stage was performed with the CRD at atmospheric pressure and with feed-water temperature regulation up to low sub-cooling. The feed-water pipeline simulated alternative configurations of the piping layout with a second injection line (dummy) to test possible interference of pulses.

The ad-hoc test loop (CEM, Circuito de Ensayo de Mecanismos, Fig. 5) was designed to allow automatic control of the flow, pressure and temperature; its instrumentation produces the information on operating parameters including pulse shape and timing. The tests included characterization of the mechanism and of the driving water circuit under different operating conditions and the study of abnormal situations such as increase in drag forces, pump failure, loss of control over water flow or temperature, saturated water injection, impact of suspended particles, and pressure "noise" in feeding line.

FIG. 5. Low Pressure Test Rig.

The tests were performed in turbulent regime, under conditions as close to the actual operating ones as it was possible to simulate in this loop, and have shown good reliability and repetitiveness as well as acceptable sensitivity margins for all relevant variables within the capability of a standard control system.

Qualification Tests

A high-pressure loop (CAPEM, Circuito de Alta Presión para Ensayo de Mecanismos) is being designed in order to reach the actual operating conditions (P = 12.25 MPa, T ≈ 326°C). The main objectives are to verify the behaviour of the mechanisms, to tune up the final controlling parameter values and to perform endurance tests. After this, the operation under abnormal conditions will be tested, such as CRD behaviour during RPV depressurisation caused by simulated breakage of feeding pipes, etc.

7. RPV INTERNALS TESTING

Mechanical structures of the core, supporting guides and of all parts of the kinematics chain of the First Shutdown System are of particular interest within this series of tests. Complex assemblies and structures like Steam Generator Units or ad-hoc mechanical solutions require the evaluation of manufacturing and assembly process before finishing the design stage.

In sum, the internals should be verified in order to define manufacturing and assembling allowances and other detailed engineering parameters in a way that would ensure compliance with the functional requirements during RPV lifetime. Most tests are performed at mock-up facilities of 1:1 vertical scale.

8. IN-VESSEL INSTRUMENTATION

Since the HCRD design adopted has no movable parts outside the RPV, it is necessary to design a special probe to measure the rod position, and this probe should be able to withstand the primary environment conditions. The proposed design consists of a coil wired around the HCRD cylinder with an external associated circuit that measures electric reluctance variations induced by the movement of a piston-shaft (made of magnetic steel) inside the cylinder.

Cold tests were performed which show that the system is capable of sensing a single-step movement of the regulating CRD with an acceptable accuracy. In-furnace high temperature tests will be conducted to evaluate behaviour of the system under temperature changes similar to those occurring during operational transients.

The design activities for special high-pressure removable feed troughs to provide a path for dozens of electrical signals through the RPV cover are underway. They include development and qualification of specific manufacturing and welding techniques.

9. TOOLS FOR ECONOMIC EVALUATION

Due to large number of parameters to be determined and complex relations that may exist between them, reactor designing becomes an intrinsically complex task. At the conceptual engineering stage, quantifying the impacts of the mechanics, thermal-hydraulics, neutron physics and safety on reactor costs is of interest. A breakdown for the main items that affect costs should be determined in order to find a unit cost of the generated energy and a figure of merit with respect to alternative designs.

Under the CAREM programme, a computational tool is being developed to solve the above-mentioned problem and, in this way, to provide a support to the design team during conceptual design stage. This code, called IREP+NS –Integrated Reactor Evaluation Program plus Nuclear Safety, performs the internal iterations necessary to obtain a coherent set of design and operational parameters that define a reactor, while also making consideration of the major feedbacks that exist between them. The code allows the designers to perform economic optimisation of the more important parameters of the core, primary system, safety systems and secondary systems in order to reduce the overall cost of electricity generation [6, 7].

IREP+NS performs neutronic, thermal hydraulic, mechanical, economic and nuclear safety evaluation of the reactor and produces the levelled electricity generation costs as basic output, while pointing to the specific outcome in different technical areas. This code makes it possible to perform optimisation of the parameters that produce maximum impact on the cost of energy generated by a reactor under consideration. Before optimisation, a series of basic engineering decisions, such as that on reactor unit power, are delegated to steady-state calculation routines. As a first step, a set of the basic design parameters that correspond to the initial design are introduced to these routines, which incorporate mechanical, thermal-hydraulic, neutronic and economic models to calculate the plant. At the next step, the results, which include the figure of merit, namely the generation cost, are inputted to the optimisation routine. Design restrictions are verified while this routine looks for a more economical design. A new set of design parameters replaces the previous one, and the process continues in iterations.

Although the current methodologies including both classical and more advanced ones, such as steady state optimisation, may ensure the fulfilment of safety-related design requirements, the lack of a balance between safety and economy is often quite obvious. Therefore, it is beneficial that economy and safety are evaluated together at a conceptual design stage in order to balance properly these two fundamental aspects of any design. A global approach to this process is important to contemplate the design feedbacks between all systems in all involved areas. Safety aspects of the design provide one of the most notable contributions to costs, hence they should be considered in a cost-effective way. As other authors have already pointed out, this new approach should involve new methods for cost-benefit and ALARA analyses employing modern PSA techniques to fulfil all basic safety requirements under realistic models and assumptions rather than produce overly detailed prescriptions for the unlimited improvement of safety.

10. CARA PROJECT

Argentina has two Pressurized Heavy Water Reactor (PHWR) NPPs in operation and one under construction. They have quite different designs, particularly as comes to fuel elements. The fuel elements were originally designed for natural uranium and had a discharge burn-up of less than 8,000 MWd/t UO_2. Later on, for Atucha-I plant this burn-up was increased up to 11,000 MWd/t UO_2 with the use of slightly enriched uranium fuel (SEU). Both nuclear power plants use on-line refuelling, but they differ in the number and length of the refuelled elements. CANDU type reactor in Embalse on the total has 12 fuel elements in 6 m long horizontal channel, of which two are refuelled at a time. Different from that, the vertical channel of Atucha-I plant has one single fuel bundle of 5.25 m active length, hanged by its upper part.

Under the CARA project [8] CNEA is developing an advanced fuel bundle concept for heavy water reactors, specially designed to fit the Argentinean fuel cycle requirements. The CARA fuel bundle can be used in the reactors of both types and will substantially improve the competitiveness of nuclear option in Argentina.

Advanced designs that consider different fuel rods and increase the number of different structural elements and related technological operations increase fabrication costs, which may be essential on a scale of small and medium populated developing countries. New advanced fuel bundle to be developed in Argentina is designed not only to increase safety margins and fuel burn-up but to reduce fabrication costs also.

The CARA fuel design is adjusted to the conditions of the operating NPPs, mainly the coolant flow and hydraulic channel pressure drop, and is mechanically compatible with the refuelling machines of the vertical and horizontal channel reactors.

The CARA fuel bundle has been designed to improve major fuel performance characteristics of the reactors of both types [9]. The CARA fuel can reach higher burn-up with the use of SEU and ensures higher thermal-hydraulic and safety margins (due to increased number of fuel rods), together with the lower fuel pellet centre temperature and Zircaloy /Heavy Metal mass ratio. Moreover, it preserves the linear mass density of fuel by using a single fuel rod diameter and by minimizing the number of welds on claddings (3 spacer grids per fuel bundle are used, which is similar to PWR technology). The “classical” spacer pads welded on the claddings are not used in this design.

CARA fuel bundle includes 52 single diameter collapsible fuel rods of about 1 meter length each, fastened by three spacer grids and two end plates. It uses SEU of 0.9% enrichment that secures the discharge burn-up of 14,500 MWd/t U. For Atucha-I it is necessary to join together five CARA bundles using an additional (external) coupling system.

The project attracted interest of the Argentinean nuclear power utility and fuel manufacturing company and has been divided in three phases, in line with the growing complexity and costs and the reduced technology development risk.

The first phase, in which two CARA fuel bundle prototypes have been built to perform hydraulic characterization of a spacer grid in low pressure drop test loop, is completed. This phase also included initial analysis of the vibrations in fuel rod - spacer spring system. Three spacer grid prototypes were tested.

The second phase is in progress currently, which makes full account of the results from the first phase. Computational models were developed and validated on experimental results for hydraulic pressure drop. Critical heat flux margins were calculated by COBRA-IV code and validated with the use of published data on critical power measurements for very different cluster geometries relevant for CARA. Computational probabilistic analysis of fuel performance was performed with the use of BACO code [4,10] and facilitated the determination of dimensional tolerances.

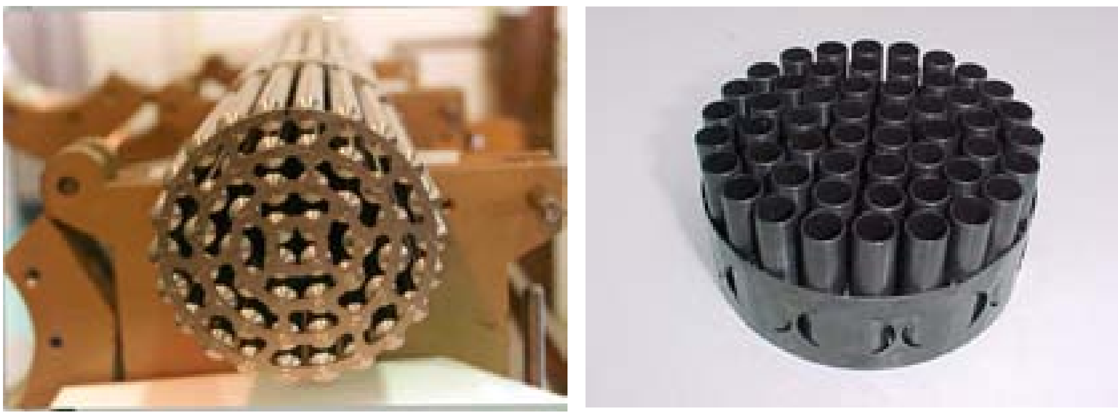

FIG. 10. CARA fuel bundle and spacer grid prototypes.

Two different coupling systems (for the Atucha-I plant) were built for hydraulic characterization. Seven CARA fuel bundle prototypes with spacer grids of new design and with the enhanced welding between end plate and fuel rods are under fabrication. Irradiation testing programme to simulate extreme irradiation conditions was developed. Initial stages of the irradiation programme will be conducted at the OECD's Halden Reactor.

Licensing application will be submitted after the completion of irradiation tests and post irradiation evaluations.

11. CONCLUSION

The RD&D programme of the CAREM project has been advanced to the stage when robustness of the design components is to be demonstrated. Development of computer tools to assist the designer in cost-benefit safety design optimisation could be rated as one of important achievements of the CAREM project.

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DESIGN AND SAFETY OF IRIS, AN INTEGRAL WATER COOLED SMR FOR NEAR TERM DEPLOYMENT

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Abstract. International Reactor Innovative and Secure (IRIS) is an integral, modular, medium size (335 MWe) PWR, geared at near term deployment in developing as well as developed countries. It has been under development since the turn of the century by an international consortium led by Westinghouse and including 21 organizations from 10 countries. Described here are its salient features, primarily its integral design (which includes steam generators, pump, pressurizer and control rod drive mechanisms inside the vessel, together with the core, control rods, and neutron reflector/shield) and the "safety-by-design"™ IRIS philosophy. This unique approach, by eliminating accidents at the design stage, or decreasing their consequences/probabilities when outright elimination is not possible, provides a very powerful first level of defence in depth. The "safety-by-design"™ allows a significant reduction and simplification of the passive safety systems, which are presented here, together with an assessment of the IRIS response to transients and postulated accidents. A brief summary is also provided of the IRIS approach to enable maintenance over a 48-month schedule, thus allowing a significant reduction in refuelling frequency. With its moderate size IRIS provides the ability to gradually add generating capacity according to market needs, at the same time reducing financial burden. Thus, IRIS is particularly well suited for small/medium size power grids.

1. INTRODUCTION

The IRIS plant conceptual design was completed in 2001 and the preliminary design is nearing completion. The currently underway pre-application licensing process with NRC started in October 2002. Details of the IRIS design and supporting analyses have been previously reported and the reader is directed to the listed references. Purpose of this paper is to provide an overall review of the IRIS characteristics [4,5].

IRIS is a pressurized water reactor that utilizes an integral reactor coolant system layout. The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant system components including pumps, steam generators, pressurizer, control rod drive mechanisms and neutron reflector. The IRIS integral vessel is larger than a traditional PWR pressure vessel, but the size of the IRIS containment is a fraction of the size of corresponding loop reactors, resulting in a significant reduction in the overall size of the reactor plant.

IRIS has been primarily focused on achieving design with innovative safety characteristics. The first line of defence in IRIS is to eliminate event initiators that could potentially lead to core damage. In IRIS, this concept is implemented through the "safety-by-design"™ IRIS philosophy, which can be simply described as "design the plant in such a way as to eliminate accidents from occurring, rather than coping with their consequences." If it is not possible to eliminate certain accidents altogether, then the design inherently reduces their consequences and/or decreases their probability of occurring. The key difference in the "safety-by-design"™ IRIS philosophy from previous practice is that the integral reactor design is conducive to eliminating accidents, to a degree impossible in conventional loop-type reactors. The elimination of the large LOCAs, since no large primary penetrations of the reactor vessel or large loop piping exist, is only the most easily visible of the safety potential characteristics of integral reactors. Many others are possible, but they must be carefully exploited through a design process that is kept focused on selecting design characteristics that are most amenable to eliminate accident-initiating events.

The IRIS design builds on the proven technology provided by over 40 years of operating PWR experience, and on the established use of passive safety features pioneered by Westinghouse in the NRC certified AP600 plant design. The use of passive safety systems provides improvements in plant simplification, safety, reliability, and investment protection over conventional plant designs. Because of the "safety-by-design"™ approach, the number and complexity of these passive safety systems and required operator actions are further minimized in IRIS. The net result is a design with significantly reduced complexity and improved operability, and extensive plant simplifications to reduce construction time.

2. THE IRIS APPROACH AND THE IRIS CONSORTIUM

When Westinghouse started the conceptual design of a new reactor in answer to the DOE solicitation, the overriding objective was to develop a commercially viable concept and thus avoid producing just one more paper reactor like so many of its predecessors. It was evident that the era of a single company, or even a single nation, developing and deploying a nuclear plant had past. Also, it was apparent that many utilities, as well as developing nations, are interested in capping their capital investment in a power plant project to only a few hundred million dollars, thus driving them to concentrate on smaller capacity additions. Larger plants, however, have economy of scale and a new dimension has to appear for smaller plants to become more economical and true market competitors.

Smaller, modular gas cooled reactors had already been proposed as commercial market entries, the Pebble Bed Modular Reactor (PBMR) [18] and the Gas Turbine-Modular Helium Reactor (GT-MHR) [13]. For the PBMR, Exelon had made a strong case of the inherent advantage of small plants in introducing new power to the grid in limited increments, thus finely tailoring supply and demand and limiting the utilities' financial exposure.

On the opposite side of the spectrum of design characteristics, small reactors offer intrinsic proliferation resistance as they can be designed to operate for long times without refuelling, thus significantly reducing access to the fuel. Recently the Agency has launched a program devoted to small/medium reactors without on-site refuelling. As mentioned, economics and "no" refuelling are somewhat antithetic, since a straight burn cycle cannot produce optimum burn-up and fuel cycle cost. Also, some reactor types are more amenable than others to very long life cores; they are fast spectrum, liquid metal cooled reactors, which past experience has however shown to be more expensive than thermal spectrum, water or gas cooled reactors.

In the initial phase of the IRIS design, particular attention was dedicated to its proliferation resistance characteristics and core designs with long fuel cycles of the order of eight years or more were developed. Because of economic considerations and utilities feedback, the reference design was eventually chosen as a shorter cycle, but retaining the capability to accept long straight-burn cores if so desired (or required).

A common feature of modular reactors is the fact that, in addition to being simpler to construct and operate, these smaller plants have to be fabricated in series. Thus, it is readily apparent that to fabricate and deploy an economically large enough number of multiple, identical modules, the market has to be one global, international arena.

Once it was established that this new reactor was to be deployed world-wide, it followed that to be readily accepted internationally, it had to be developed internationally, i.e., it had to address international requirements, needs and even cultures. Hence, the IRIS approach, as emphasized by the first letter (International) of its acronym: From the very beginning, IRIS was going to be designed and subsequently fabricated, deployed and serviced by an international partnership, where all team members were stakeholders in the project.

This approach immediately found a positive resonance, as the IRIS team kept growing in its first 3 years from the initial 4 members and 2 countries to the present 21 members from 10 countries. The original team included Westinghouse, two American universities (University of California Berkeley and Massachusetts Institute of Technology (MIT)) and one Italian university (Polytechnic of Milan). Later on other reactor designers and component manufacturers, fuel and fuel cycle vendors, architect engineers, power producers, universities, and laboratories joined the team. Table 1 provides a summary of the IRIS team partnership with the areas of responsibility of each team member. Associate members are US universities and laboratories working on DOE funded Nuclear Energy Research Initiative (NERI) projects, which, while of general interest, use IRIS as the example application of the technology being investigated.

While associated members are DOE funded via NERI, all other IRIS consortium members (including international universities) are currently self-funded and provide to the project both design effort and previous know-how. Currently, approximately 100 people across the IRIS consortium are contributing to the project.

The contribution of the universities to the IRIS program cannot be emphasized enough. Innovative design solutions have been proposed and developed by universities, and IRIS is perhaps the first and only commercial reactor project where academia and industry are in a partnership equally co-responsible for the design. The partnership with universities (and laboratories) has also a potentially very important long-term effect, in making IRIS a “living and contemporary” design. In fact, once the IRIS preliminary design is completed, its implementation becomes essentially the responsibility of the industrial partners, while the universities and laboratories will shift to work on future improved designs incorporating the most recent technological advancements. As they are ready, industry can then implement them in a new series of IRIS modules. A key reason that this can conceivably be done and accepted by the market is that the size of an IRIS module is only about one-third to one-fourth of today’s large light water reactors (LWRs) and thus the financial exposure is much more limited.

3. THE INTEGRAL REACTOR COOLANT SYSTEM

The IRIS reactor vessel (RV) [7] houses not only the nuclear fuel and control rods, but also all the major reactor coolant system (RCS) components (see Fig. 1): eight small, spool type, reactor coolant pumps (RCPs); eight modular, helical coil, once through steam generators (SGs); a pressurizer located in the RV upper head; the control rod drive mechanisms (CRDMs); and, a steel reflector which surrounds the core and improves neutron economy, as well as it provides additional internal shielding. This integral RV arrangement eliminates the individual component pressure vessels and large connecting loop piping between them, resulting in a more compact configuration and in the elimination of the large loss-of-coolant accident as a design basis event.

TABLE 1. MEMBER ORGANIZATIONS OF THE IRIS CONSORTIUM

INDUSTRY		
Westinghouse	USA	Overall coordination; leading core design, safety analyses and licensing
BNFL	UK	Commercialisation and fuel cycle
Ansaldo Energia	Italy	Steam generators design
Ansaldo Camozzi	Italy	Steam generators fabrication
ENSA	Spain	Pressure vessel and internals
NUCLEP	Brazil	Containment
Bechtel	USA	BOP, AE
OKBM	Russia	Testing, desalination and district heating co-generation
LABORATORIES		
ORNL	USA	I&C, PRA, desalination, shielding, pressurizer
CNEN	Brazil	Transient and safety analyses, pressurizer, desalination
ININ	Mexico	PRA, neutronics support
LEI	Lithuania	Safety analyses, PRA, district heating co-generation
UNIVERSITIES		
Polytechnic of Milan	Italy	Safety analyses, shielding, thermal hydraulics, steam generators design, advanced control system
MIT	USA	Advanced cores, maintenance
Tokyo Inst. of Technology	Japan	Advanced cores, PRA
University of Zagreb	Croatia	Neutronics, safety analyses
University of Pisa	Italy	Containment analyses, severe accident analyses, neutronics
Polytechnic of Turin	Italy	Source term
University of Rome	Italy	Radioactive waste system, occupational doses
POWER PRODUCERS		
TVA	USA	Maintenance, utility perspective
Eletronuclear	Brazil	Developing country utility perspective
ASSOCIATED US UNIVERSITIES (NERI PROGRAMS)		
University of California Berkeley	USA	Neutronics, advanced cores
University of Tennessee	USA	Modularisation, I&C
Ohio State University	USA	In-core power monitor, advanced diagnostics
Iowa State University (and Ames Lab)	USA	On-line monitoring
University of Michigan (and Sandia Labs)	USA	Monitoring and control

As the IRIS integral vessel contains all the RCS components, it is larger than the RV of a traditional loop-type PWR. It has an inner diameter of 6.21 m and an overall height of 22.2 m including the closure head. Water flows upwards through the core and then through the riser region (defined by the extended core barrel). At the top of the riser, the coolant is directed into the upper part of the annular plenum between the extended core barrel and the RV inside wall, where the suction of the reactor coolant pumps is located. Eight coolant pumps are employed, and the flow from each pump is directed downward through its associated helical coil steam generator module. The primary flow path continues down through the annular downcomer region outside the core to the lower plenum and then back to the core completing the circuit.

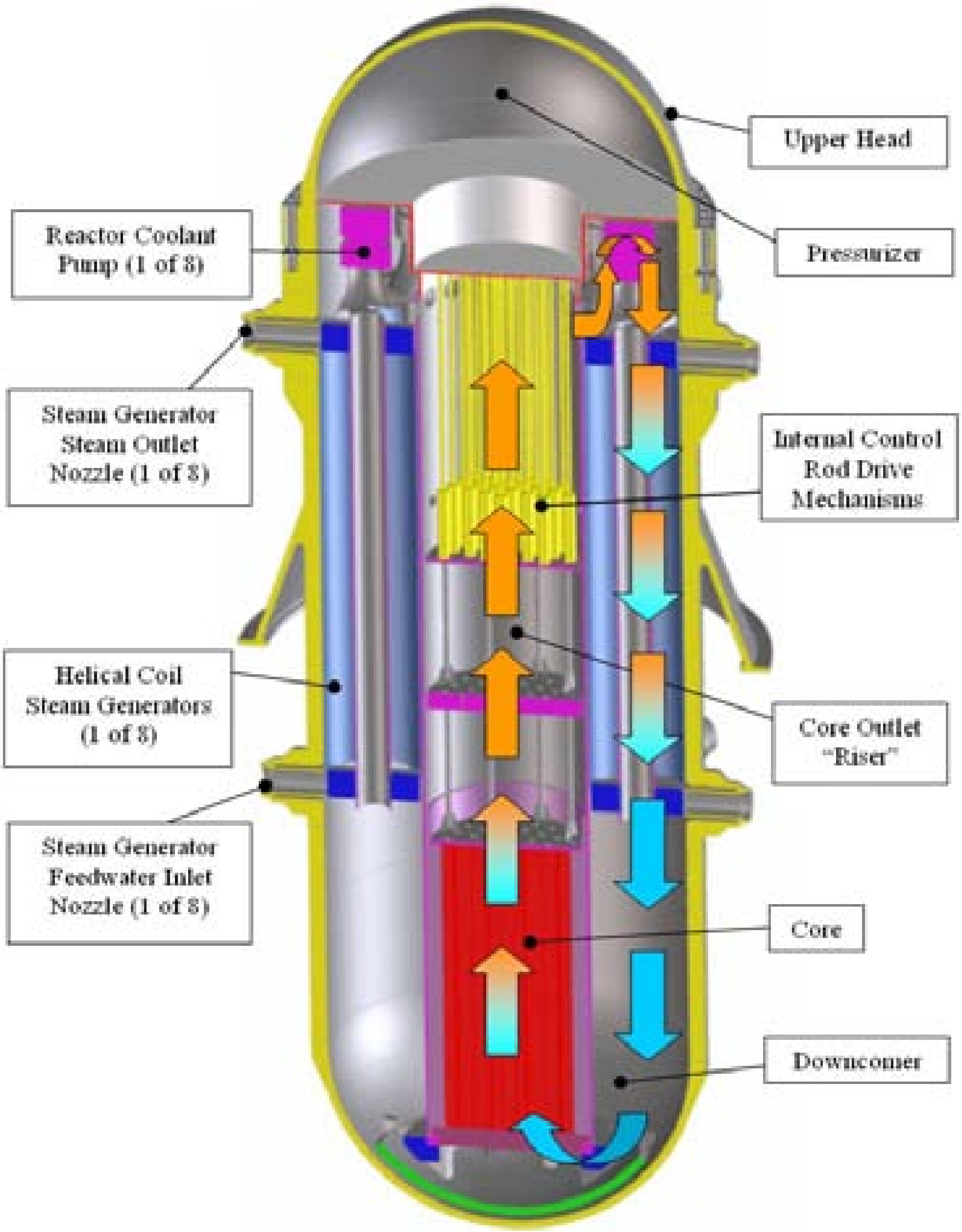

FIG. 1. IRIS integral layout.

The major in-vessel components are described below:

- **Pressurizer** – The IRIS pressurizer [1] is integrated into the upper head of the reactor vessel (see Fig. 2). The pressurizer region is defined by an insulated, inverted top-hat structure that divides the circulating reactor coolant flow path from the saturated pressurizer water. This structure includes a closed cell insulation to minimize the heat transfer between the hotter pressurizer fluid and the subcooled primary water. Annular heater rods are located in the bottom portion of the inverted top hat, which contains holes to allow water insurge and outsurge to/from the pressurizer region. These surge holes are located just below the heater rods so that insurge fluid flows up along the heater elements.

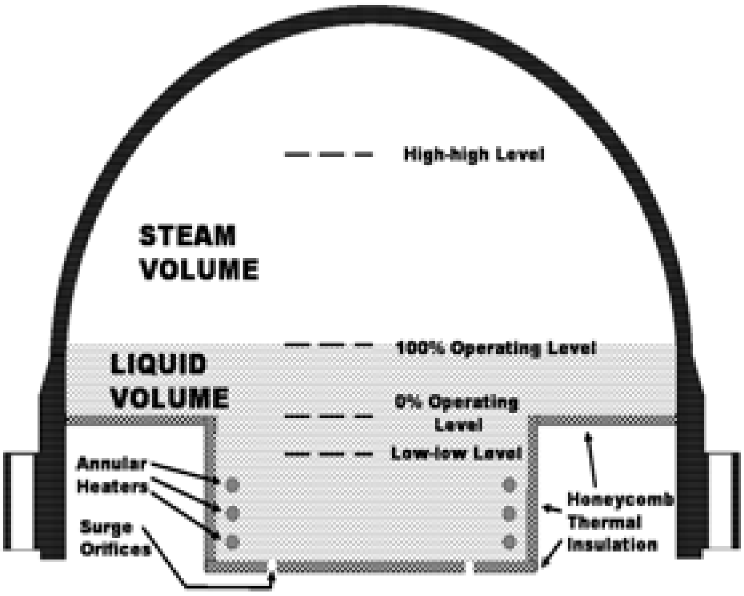

FIG. 2. IRIS pressurizer.

By utilizing the upper head region of the reactor vessel, the IRIS pressurizer provides very large water and steam volume, as compared to plants with a traditional, separate, pressurizer vessel. The IRIS pressurizer has a total volume of ~71 m^3, which includes a steam volume of ~49 m^3. The steam volume is about 1.6 times bigger than the AP1000 pressurizer steam space, while IRIS has less than 1/3 the core power. The large steam volume to power ratio is a key reason why IRIS does not require pressurizer sprays, which are used in current PWRs to prevent the pressurizer safety valves from lifting for any design basis heatup transients.

- **Reactor core** –The IRIS core (Fig. 3) and fuel assemblies are similar to those of a loop type Westinghouse PWR design. Specifically, the IRIS fuel assembly design is similar to the Westinghouse 17×17 XL Robust Fuel Assembly design and AP1000 fuel assembly design. An IRIS fuel assembly consists of 264 fuel rods with a 0.374-inch outer diameter in a 17×17 square array. The central position is reserved for in-core instrumentation, and 24 positions have guide thimbles for the control rodlets. Low-power density is achieved by employing a core configuration consisting of 89 fuel assemblies with a 14-ft (4.267 m) active fuel height, and a nominal thermal power of 1000 MWth. The resulting average linear power density is about 75% of the AP600 value. The improved thermal margin provides increased operational flexibility, while enabling longer fuel cycles and thus increased overall plant capacity factors.

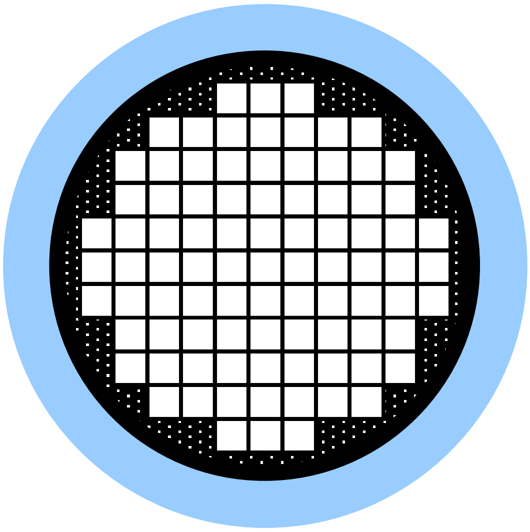

FIG. 3. IRIS core and neutron reflector.

The initial, reference IRIS core [20] will use UO_2 fuel, enriched to 4.95 w/o in ^{235}U, with lower enrichment in the axial blankets and at the core periphery. The fission gas plenum length is increased (roughly doubled) compared to current PWRs, thus eliminating potential concerns with internal overpressure. The integral RV design permits this increase in the gas plenum length with practically no penalty, because the steam generators mainly determine the vessel height. The 89 assembly core configuration has a relatively high fill-factor (i.e. it closely approximates a cylinder), to minimize the vessel diameter.

Reactivity control is accomplished through solid burnable absorbers, control rods, and the use of a limited amount of soluble boron in the reactor coolant. The reduced use of soluble boron makes the moderator temperature coefficient more negative, thus increasing inherent safety. The initial core is designed for a three- to three-and-half-year cycle with half-core reload to optimise the overall fuel economics while maximizing the discharge burn-up. In addition, a four-year straight burn fuel cycle can also be implemented to improve the overall plant availability, but at the expense of a somewhat reduced discharge burn-up.

Also, as previously mentioned, IRIS core designs capable of 8-year straight burn cycle have been developed using UO_2 or MOX fuel with fissile content increased to 7-10% [21]. This is facilitated by the "variable moderation approach", summarized in Table 2, whereas the moderator-to-fuel ratio is increased with the increased fissile content, to achieve adequate neutron thermalisation.

TABLE 2. VARIABLE MODERATION APPROACH

	Reference Core	Future UO_2 Upgrade	Future MOX Upgrade
Fuel Type	UO_2 <5% fissile	UO_2 >5% fissile	MOX >5% fissile
Fissile Content	4.95%	~7-8%	~9-10%
Core Lifetime (straight burn)	~4 years	~8 years	~8 years
P/d	1.4	1.45	1.7
Vm/Vf	2.0	2.2	3.7

- **Reactor coolant pumps** – The IRIS RCPs [12] are of a "spool type," which has been used in marine applications, and are being designed and will soon be supplied for chemical plant applications requiring high flow rates and low developed head. The motor and pump consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor that carries high specific speed pump impellers. The spool type pump is located entirely within the reactor vessel, with only small penetrations for the electrical power cables and for water-cooling supply and return. Further, significant qualification work has been completed on the use of high temperature motor windings. This and continued work on the bearing materials has the potential to eliminate even the need for cooling water and the associated piping penetrations through the RV. This pump compares very favourably to the typical canned motor RCPs, which have the pump/impeller extending through a large opening in the pressure boundary with the motor outside the RV. Consequently, the canned pump motor casing becomes part of the pressure boundary and is typically flanged and seal welded to the mating RV pressure boundary surface. All of this is eliminated in IRIS. In addition to the above advantages derived from its integral location, the spool pump geometric configuration maximizes the rotating inertia and these pumps have a high run-out flow capability. Both these attributes mitigate the consequences of loss-of-flow accidents (LOFAs). Because of their low developed head, spool pumps have never been candidates for nuclear applications. However, the IRIS integral RV configuration and low primary coolant pressure drop can accommodate these pumps and together with the assembly design conditions can take full advantage of their unique characteristics.

- **Steam generators** – The IRIS SGs are once-through, helical-coil tube bundle design with the primary fluid outside the tubes [6]. As shown in Fig. 4, eight steam generator modules are located in the annular space between the core barrel (outside diameter 2.85 m) and the reactor vessel (inside diameter 6.21 m).

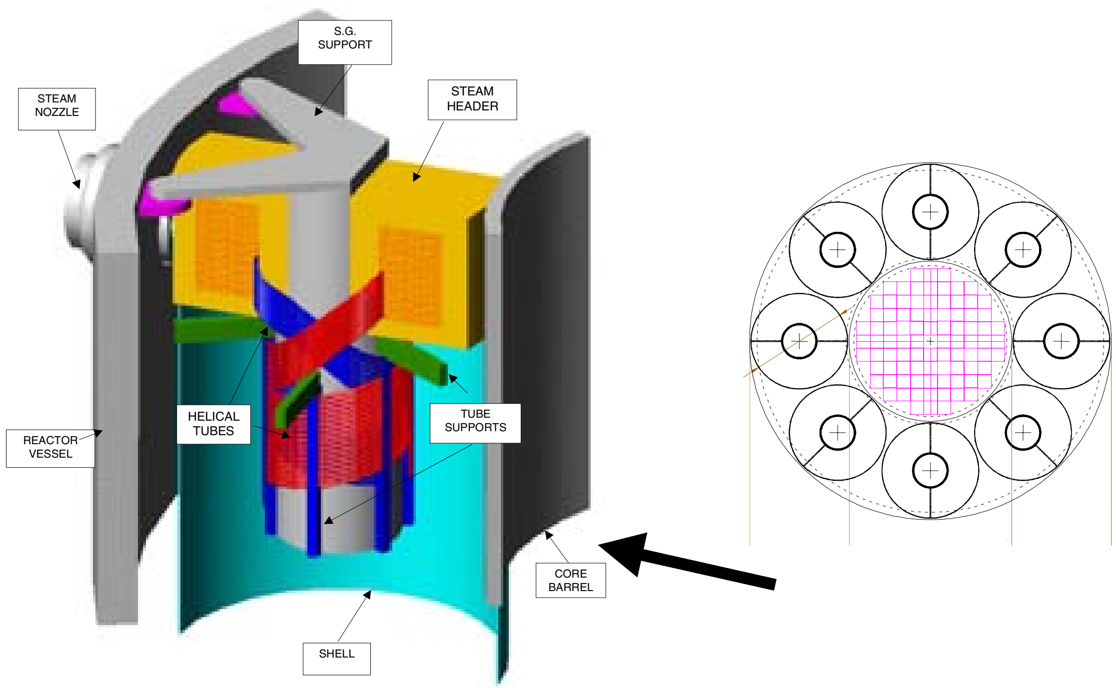

FIG. 4. IRIS helical coil steam generator.

Each IRIS SG module consists of a central inner column, which supports the tubes, the lower feed water header and the upper steam header. The enveloping outer diameter of the tube bundle is 1.64 m. Each SG has 656 tubes, and the tubes and headers are designed for the full external RCS pressure. The tubes are connected to the vertical sides of the lower feedwater header and the upper steam header. The SG is supported from the RV wall and the headers are bolted to the vessel from the inside of the feed inlet and steam outlet pipes. Figure 4 illustrates the IRIS helical coil SG upper steam discharge header and the tube bundle arrangement. The helical-coil tube bundle design is capable of accommodating thermal expansion without excessive mechanical stress, and has high resistance to flow-induced vibrations. Ansaldo tested a prototype of this SG successfully in an extensive test campaign conducted on a 20 MWth full diameter, part height test article. The performance characteristics (thermal, vibration, pressure losses) were investigated along with the determination of the operating characteristics domain for stable operation.

♦ **Control rod drive mechanisms** – The integral configuration is ideal for locating the CRDMs inside the vessel, in the region above the core and surrounded by the steam generators. Their advantages are in safety and operation.

Safety-wise, the uncontrolled rod ejection accident (a Class IV accident) is eliminated because there is no potential differential pressure in excess of 2000 psi to drive out the CRDM extension shafts. Operation-wise, the absence of CRDM nozzle penetrations in the upper head eliminates all the operational problems related with corrosion cracking of these nozzle welds and seals which have intermittently plagued the industry, and most recently have extensively flared up (e.g., the Davis-Besse plant). The design and manufacturing of the upper head is also simpler and cheaper. Integral reactor designs featuring internal CRDMs were small, low power, like the Argentinean CAREM [14] and the Chinese NHR [2,10], which employ hydraulically driven rods, and the Japanese MRX [12], which uses an electromagnetic drive mechanism. Very recently, however, they have been proposed in Japan for large BWRs [17].

Thus, IRIS has adopted the internal CRDMs as reference (traditional CRDMs remaining as backup) because (1) they eliminate the corrosion problem, (2) they are one more implementation of the safety-by-design™ IRIS philosophy, and (3) current advancements which have occurred in Japan in regard to the electromagnetic concept, while internally to the IRIS project, Polytechnic of Milan has further advanced the hydraulic drive concept. IRIS is currently evaluating candidate concepts for the internal CRDMs to proceed with the preliminary design of the chosen one.

♦ **Neutron reflector** – IRIS features a stainless steel radial neutron reflector to lower fuel cycle cost and to extend reactor life. This reflector reduces neutron leakage thereby improving core neutron utilization, and enabling extended fuel cycle and increased discharge burn-up. The radial reflector has the added benefit of reducing the fast neutron fluence on the core barrel, and, together with the thick downcomer region, it significantly reduces the fast neutron fluence on the reactor vessel as well as the dose outside the vessel to the extent of yielding, for any practical purposes, a “cold” vessel. This has obvious beneficial impacts on costs (very long life vessel, no need for the embrittlement surveillance program, reduced biological shield), operational doses, and decommissioning.

4. EXTENDED MAINTENANCE

As mentioned, a distinguishing characteristic of IRIS is its capability of operating with long cycles. Even though the reference design features a two-batch, three-year fuel cycle, selected on the basis of ease of licensing and U.S. utilities preference, IRIS is capable of eventually operating in straight burn with a core lifetime of up to eight years. However, the significant advantages connected with a long refuelling period in reducing operation and maintenance (O&M) costs are lost if the reactor still has to be shut down on a 18 to 24-month interval for routine maintenance and inspection. Thus, first and foremost, the IRIS primary system components are designed to have very high reliability to decrease the incidence of equipment failures and reduce the frequency of required inspections or repairs. Next, IRIS has been designed to extend the need for scheduled maintenance outages to at least 48 months. The basis of the design has been a study [15] performed earlier by MIT for an operating PWR to identify required actions for extending the maintenance period from 18 to 48 months. The strategy was to either extend the maintenance/testing items to 48 months or to perform maintenance/testing on line. MIT identified 3743 maintenance items, 2537 of them off-line and the remaining 1206 on-line. It was also confirmed that 1858 of the off-line items could be extended from 18 to 48 months, while 625 could be re-categorized from off-line to on-line. Further, out of the 1858 items there were 1499, which were electrical surveillances and had a strong potential for also being performed on-line. This left only 54 items, which still needed to be performed off-line on a schedule shorter than 48 months. Starting from this MIT study and factoring in the specific IRIS conditions (for example, there is no need to change the RCP oil lubricant, since the spool type pumps are lubricated by the reactor coolant), only 7 items were left as obstacles to a 48-month cycle [8]. These items have been addressed and either have been resolved or a plan of action has been identified [3]

Because of the four-year maintenance cycle capability, the capacity factor of IRIS is expected to comfortably satisfy and exceed the 95% target and it is expected that personnel requirements will be significantly reduced. Both considerations will result in decreased O&M costs.

Uninterrupted operation for 48 months requires reliable advanced diagnostics. The IRIS project is currently investigating various technologies, either already proven or in advanced phase of development, to monitor the behaviour of the in-vessel components. Promising, but more distant technologies, are being pursued by associated universities.

5. CONTAINMENT SYSTEM

Because the IRIS integral RV configuration eliminates the loop piping and the externally located steam generators, pumps and pressurizer with their individual vessels, the footprint of the patent-pending IRIS containment system is greatly reduced. This size reduction, combined with the spherical geometry, results in a design pressure capability at least three times higher than a typical loop reactor cylindrical containment, assuming the same metal thickness and stress level in the shell. The current layout features a spherical, steel containment vessel (CV) that is 25 m (82 ft.) in diameter (see Fig. 5).

The CV is constructed of 1-¾ in. steel plate and has a design pressure capability of 1.4 MPa (~190 psig). The containment vessel has a bolted and flanged closure head at the top that provides access to the RV upper head flange and bolting. Removing the containment vessel closure head, installing a sealing collar between the CV and RV, and removing the RV head accomplish refuelling of the reactor. The refuelling cavity above the containment and RV is then flooded, and the RV internals are removed and stored in the refuelling cavity. Fuel assemblies are vertically lifted from the RV directly into a fuel handling and storage area, using a refuelling machine located directly above the CV. Thus, no refuelling equipment is required inside containment and the single refuelling machine is used for all fuel movement activities.

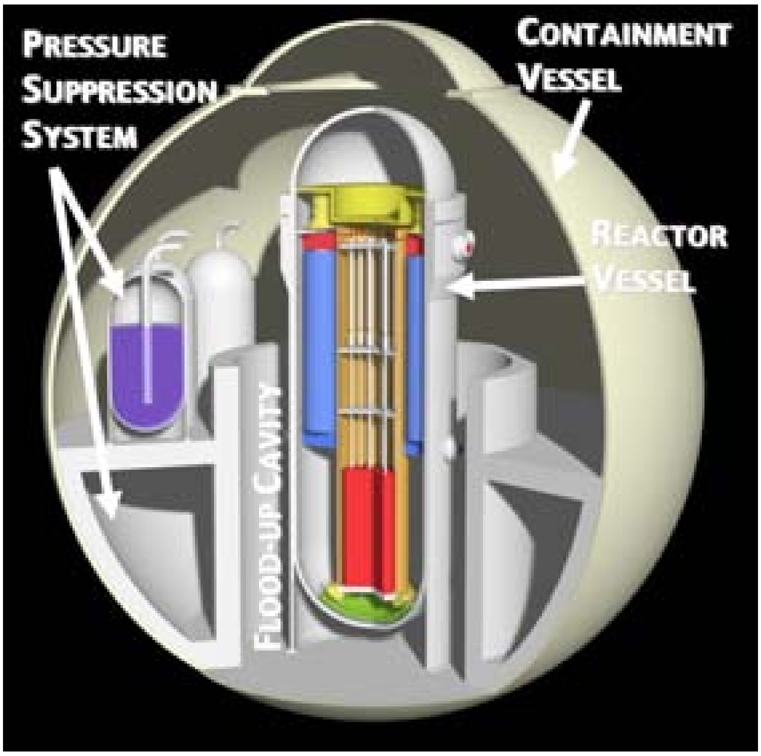

FIG. 5. IRIS spherical steel containment arrangement.

Figure 5 also shows the pressure suppression pool that limits the containment peak pressure to well below the CV design pressure. The suppression pool water is elevated such that it provides a potential source of gravity driven makeup water to the RV. Also shown is the RV flood-up cavity formed by the containment internal structure. The flood-up level is 9 m and ensures that the lower section of the RV, where the core is located, is surrounded by water following any postulated accident. The water flood-up height is sufficient to provide long-term gravity makeup, so that the RV water inventory is maintained above the core for an indefinite period of time. It also provides sufficient heat removal from the external RV surface to prevent any vessel failure following beyond design basis scenarios.

Almost half of the IRIS containment vessel is located below ground, thus leaving only about 15 m above the ground (i.e., several times less than the containment of a large LWR). This very low profile makes IRIS an extremely difficult target for aircraft flying terrorists; in addition, the IRIS containment is inconspicuously housed in and protected by the reactor building. The cost of putting the entire reactor underground was evaluated; it was judged to be prohibitive for a competitive entry to the power market and unnecessary since the IRIS design characteristics are such to offer both an economic and very effective approach to this problem.

6. THE SAFETY-BY-DESIGN™ IRIS PHILOSOPHY

The IRIS design provides for multiple levels of defence for accident mitigation (in-depth-in-depth (DID)), resulting in extremely low core damage probabilities. In addition to the traditional DID levels (barriers, redundancy, diversity, etc.) IRIS introduces a very basic level of DID, i.e., elimination by design of accident initiators or reduction of their consequences/probability. This is implemented through the "safety-by-design" ™ IRIS philosophy, which was briefly presented in the introduction.

Several features of the design form the basis of the safety-by-design™ approach. These features are summarized in Table 3 and are discussed in the following. Table 4 provides an overview of how the safety-by-design™ features listed in Table 3 will impact the typical design basis events.

TABLE 3. IMPLICATIONS OF SAFETY-BY-DESIGN™ IRIS PHILOSOPHY

IRIS design characteristic	Safety implication	Accidents affected
Integral layout	No large primary piping	-LOCAs
Large, tall vessel	Increased water inventory Increased natural circulation Can accommodate internal CRDMs	-LOCAs -Decrease in heat removal -Various events -RCCA ejection, eliminate head penetrations
Heat removal from inside the vessel	Depressurises primary system by condensation and not by loss of mass Effective heat removal by SG/EHRS	-LOCAs -LOCAs -All events for which effective cool-down is required
Reduced size, higher design-pressure containment	Reduced driving force through primary opening	-LOCAs
Multiple coolant pumps	Decreased importance of single pump failure	Locked rotor, shaft seizure/break
High design-pressure steam generator system	No SG safety valves Primary system cannot over-pressure secondary system Feed/steam system piping designed for full RCS pressure reduces piping failure probability	-Steam generator tube rupture -Steam line break -Feed line break
Once-through steam generator	Limited water inventory	-Steam line break -{Feed line break}*
Integral pressurizer	Large pressurizer volume/reactor power	-Overheating events, including feed line break -ATWS

* The only accident which is potentially affected in a negative way

TABLE 4. IRIS RESPONSE TO PWR CLASS IV EVENTS

Class IV design basis event		IRIS design characteristic	Results of safety-by-design™ IRIS philosophy
1	Large break LOCA	Integral RV layout – no loop piping	Eliminated by design
2	Steam generator tube rupture	High design pressure once-through SGs, piping, and isolation valves	Reduced consequences, simplified mitigation
3	Steam system piping failure	High design pressure SGs, piping, and isolation valves. SGs have small water inventory	Reduced probability, reduced (limited containment effect, limited cooldown) or eliminated (no potential for return to critical power) consequences
4	Feedwater system pipe break	High design pressure SGs, piping, and isolation valves. Integral RV has large primary water heat capacity.	Reduced probability, reduced consequences (no high pressure relief from reactor coolant system)
5	Reactor coolant pump shaft break	Spool pumps have no shaft	Eliminated by design
6	Reactor coolant pump seizure	No DNB for failure of 1 out of 8 RCPs	Reduced consequences
7	Spectrum of RCCA ejection accidents	With internal CRDMs there is no ejection driving force	Eliminated by design
8	Design basis fuel handling accidents	No IRIS specific design feature	No impact

The adoption of an integral reactor coolant system eliminates the large loop piping required for other designs, and thus the potential for postulated large loss of coolant accidents is eliminated by design. The elimination of large break LOCAs is only the most evident safety-by-design™ feature of IRIS; others are presented here as they are a fundamental part of the IRIS in-depth in depth.

The adoption of an integral layout requires the design of a large vessel compared to other PWRs, with a tall riser above the core to allow sufficient space for the placements of the steam generators and reactor coolant pumps in the pressure vessel. This provides a large coolant inventory in the reactor coolant system, that is the basis of the IRIS response to small and medium break LOCAs, i.e., to rely on “maintaining water inventory” rather than “providing coolant injection.” Also, the large coolant inventory provides a large heat sink that acts to effectively mitigate cooldown and heatup events.

The tall riser and the reduced pressure losses in the reactor coolant system yield a large natural circulation ratio. This provides an effective circulation of coolant in the reactor coolant system to remove decay heat from the core. Finally, the tall riser provides sufficient space to accommodate internal CRDMs. Not only this allows eliminating the potential for a rod control cluster assembly (RCCA) ejection, but it also allows eliminating the CRDMs penetrations through the vessel upper head. Thus, the operational concerns associated with boron-induced corrosion of the vessel head nozzles (which have idled the Davis-Besse power station for almost two years) are eliminated by design.

Another IRIS specific feature that has been used to inherently mitigate the consequences of postulated events is the location of the steam generators inside the pressure vessel. Coupled with the large inventory, this is a fundamental feature to shape the IRIS response to postulated small and medium break LOCAs. The large heat surface available inside the vessel is used to remove the heat produced in the core during the event, and provides a mean for depressurizing the reactor coolant system by condensing inside the vessel the steam produced, as opposite to a depressurization system that relies on mass loss outside the vessel. Thus, coolant inventory is maintained. Also, the effective heat removal through the steam generators and the emergency heat removal system (see Section 6.1) provide effective mitigation for all the events that require safety grade decay heat removal.

As discussed in Section 5, the adoption of an integral layout provides an overall reduction in the dimensions of the reactor coolant system, and thus allows designing a compact, higher design pressure containment system. During the initial phases of a loss of coolant accident, the pressure in the IRIS containment is allowed to increase early in the accident, and thus the higher backpressure provides an inherent limitation to the inventory loss from the reactor coolant system. This goes hand-in-hand with the previously discussed depressurization inside the vessel, effectively and quickly zeroing the differential pressure across the break and thus terminating the small/medium LOCA. The core remains safely covered without any water makeup or injection. It should be noted that a large margin (almost 30%) to the containment design pressure is provided for all design basis accidents, and that the effective reactor coolant system and containment cooling provided by the emergency heat removal system (EHRS) rapidly reduces the pressure in the containment to minimize containment leakage following a postulated LOCA.

The IRIS once-through steam generators, with the primary coolant on the shell side provide a reduced volume of the secondary side, and this allows designing the IRIS steam system up to the isolation valves for full reactor coolant system design pressure. This in turn allows eliminating the steam generator safety valves, since the steam system is protected by the reactor coolant system safety valves; prevents the reactor coolant system from overpressurizing the steam system; and reduces the probability for piping failures since the steam and feed lines are designed for full pressure. These features play an important role in the mitigation of both the probability and the consequences of postulated steam generator tubes ruptures. Not only the potential for failures is reduced since the tubes are mostly in compression (primary coolant on the shell side), but also failure propagation is highly improbable due to tube collapse. Additionally, simply isolating the faulted steam generator provides an effective mitigation.

Another feature of IRIS steam generators is the limited available water inventory: while it limits the consequences of cooldown events, this feature also limits the available inventory in the steam generators to mitigate heatup events, like a feed line break. However, other IRIS design features, and in particular the large primary coolant inventory, more than compensate for this drawback. Also, the rapid loss of mass from the steam generators provides a means for rapid detection of the fault and thus for a rapid actuation of the safety features.

An effective means for mitigating the consequences of heatup events is provided by another IRIS design characteristic of the integral layout. A large volume is available in the reactor vessel head for the pressurizer, which is thus designed with a large steam volume, to provide an inherent mitigation to events causing a pressurization of the reactor coolant system. Not only this allows to simplify the design (IRIS does not feature a spray system nor automatic power-operated relief valves), but it also provides an inherent protection against reactor coolant system overpressurization.

6.1. IRIS Safety Features

To complement its safety-by-design™, IRIS features limited and simplified passive systems as shown in Fig. 6. They include:

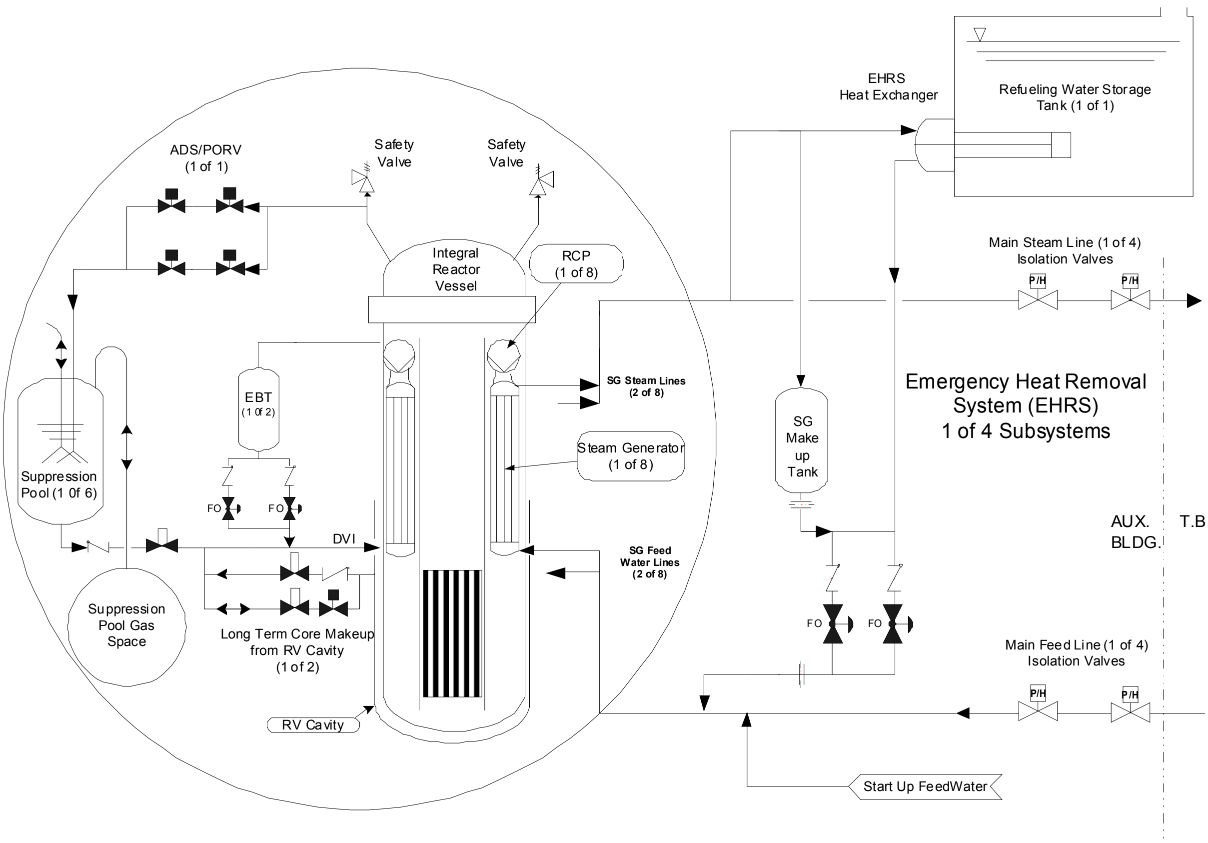

FIG. 6. IRIS passive safety system schematic.

- A passive emergency heat removal system made of four independent subsystems, each of which has a horizontal, U-tube heat exchanger connected to a separate SG feed/steam line. These heat exchangers are immersed in the refuelling water storage tank (RWST) located outside the containment structure. The RWST water provides the heat sink to the environment for the EHRS heat exchangers. The EHRS is sized so that a single subsystem can provide core decay heat removal in the case of a loss of secondary system heat removal capability. The EHRS operates in natural circulation, removing heat from the primary system through the steam generators heat transfer surface, condensing the steam produced in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. The EHRS provides both the main post-LOCA depressurization (depressurization without loss of mass) of the primary system and the core cooling functions. It performs these functions by condensing the steam produced by the core directly inside the reactor vessel. This minimizes the break flow and actually reverses it for a portion of the LOCA response, while transferring the decay heat to the environment.

- Two full-system pressure emergency boration tanks (EBTs) to provide a diverse means of reactor shutdown by delivering borated water to the RV through the direct vessel injection (DVI) lines. By their operation these tanks also provide a limited gravity feed makeup water to the primary system.

- A small automatic depressurization system (ADS) from the pressurizer steam space, which assists the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory drops below a specific level. This ADS has one stage and consist of two parallel 4 in. lines, each with two normally closed valves. The single ADS line downstream of the closed valves discharges into the pressure suppression system pool tanks through a sparger. This ADS function ensures that the reactor vessel and containment pressures are equalized in a timely manner, limiting the loss of coolant and thus preventing core uncover following postulated LOCAs even at low RV elevations.

- A containment pressure suppression system (PSS), which consists of six water tanks and a common tank for non-condensable gas storage. Each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger so that steam released in the containment following a loss of coolant or steam/feed line break accident is condensed. The suppression system limits the peak containment pressure, following the most limiting blowdown event, to less than 1.0 MPa (130 psig), which is much lower than the containment design pressure. The suppression system water tanks also provide an elevated source of water that is available for gravity injection into the reactor vessel through the DVI lines in the event of a LOCA.

- A specially constructed lower containment volume that collects the liquid break flow, as well as any condensate from the containment, in a cavity where the reactor vessel is located. Following a LOCA, the cavity floods above the core level, creating a gravity head of water sufficient to provide coolant makeup to the reactor vessel through the DVI lines. This cavity also assures that the lower outside portion of the RV surface is or can be wetted following postulated core damage events.

As in the AP600/AP1000, the IRIS safety system design uses gravitational forces instead of active components such as pumps, fan coolers or sprays and their supporting systems.

The safety strategy of IRIS provides a diverse means of core shutdown by makeup of borated water from the EBT in addition to the control rods; also, the EHRS provides a means of core cooling and heat removal to the environment in the event that normally available active systems are not available. In the event of a significant loss of primary-side water inventory, the primary line of defence for IRIS is represented by the large coolant inventory in the reactor vessel and the fact that EHRS operation limits the loss of mass, thus maintaining a sufficient inventory in the primary system and guaranteeing that the core will remain covered for all postulated events. The EBT is capable of providing some primary system injection at high pressure, but this is not necessary, since the IRIS strategy relies on “maintaining” coolant inventory, rather than “injecting” makeup water. This strategy is sufficient to ensure that the core remains covered with water for an extended period of time (days and possibly weeks). Thus, IRIS does not require and does not have the high capacity, safety grade, and high pressure safety injection system characteristic of loop reactors.

Of course, when the reactor vessel is depressurised to near containment pressure, gravity flow from the suppression system and from the flooded reactor cavity will maintain the RV coolant inventory for an unlimited period of time. However, this function would not be strictly necessary for any reasonable recovery period since the core decay heat is removed directly by condensing steam inside the pressure vessel, thus preventing any primary water from leaving the pressure vessel.

The IRIS design also includes a second means of core cooling via containment cooling, since the vessel and containment become thermodynamically coupled once a break occurs. Should cooling via the EHRS be defeated, direct cooling of the containment outer surface is provided and containment pressurization is limited to less than its design pressure. This cooling plus multiple means of providing gravity driven makeup to the core provide a means of preventing core damage and ensuring containment integrity and heat removal to the environment that is diverse from the EHRS operation.

IRIS is designed to provide in-vessel retention of core debris following severe accidents by assuring that the vessel is depressurised, and by cooling the outside vessel surface. Containing the lower part of the vessel within a cavity that always will be flooded following any event that jeopardizes core cooling cools the reactor vessel. Also, like in AP1000, the vessel is covered with standoff insulation that forms an annular flow path between the insulation and the vessel outer surface. Following an accident, water from the flooded cavity fills the annular space and submerges and cools the bottom head and lower sidewalls of the vessel [22]. A natural circulation flow path is established, with heated water and steam flowing upwards along the vessel surface, and single-phase water returning downward along the outside of the vessel insulation, to the bottom of the flood-up cavity. AP1000 testing has demonstrated that this natural circulation flow is sufficient to prevent corium melt-through. Application of AP1000 conditions to IRIS is conservative, due to the IRIS much lower core power to vessel surface ratio. The design features of the containment ensure flooding of the vessel cavity region during accidents and submerging the reactor vessel lower head in water since the liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The IRIS design also includes a provision for draining part of the water in the PSS water tanks directly into the reactor cavity.

6.2. Assessment of the IRIS Response to Transients and Postulated Design Basis Accidents

The safety-by-design™ features of the reactor, with their vastly enhanced defence in depth provide an effective means of satisfying regulatory requirements for design basis events. The main effects of this approach on IRIS safety were listed in Tables 3 and 4 and are discussed here in some detail. All the events that are typically studied as part of Section 15 of the Safety Analysis Report according to the NRC Standard Review Plan (SRP) [19], and for which IRIS will present significant differences from current active and passive PWRs, are briefly discussed here.

- **Loss of coolant accidents (LOCAs)** – The integral RV eliminates by design the possibility of large break LOCAs, since no large primary system piping is present in the reactor coolant system. Also, the probability and consequences of small break LOCA are lessened because of the drastic reduction in overall piping length, and by limiting the largest primary vessel penetration to a diameter of less than 4 in. The innovative strategy developed to cope with a postulated small break LOCA by fully exploiting the IRIS design characteristics is discussed in the following.

  IRIS is designed to limit the loss of coolant from the vessel rather than relying on active or passive systems to inject water into the RV. This is accomplished by taking advantage of the following three features of the design:

  1. The initial large coolant inventory in the reactor vessel.
  2. The EHRS, which removes heat directly from inside the RV thus depressurizing the RV by condensing steam, rather than depressurizing by discharging mass.
  3. The compact, small diameter, high design pressure containment that assists in limiting the blowdown from the RV by providing a higher backpressure in the initial stages of the accident and thus rapidly equalizing the vessel and containment pressures.

  After the LOCA initiation, the RV depressurises and loses mass to the CV causing the CV pressure to rise (blowdown phase). The mitigation sequence is initiated with the reactor trip and pump trip; the EBTs are actuated to provide boration; the EHRS is actuated to depressurise the primary system by condensing steam on the steam generators (depressurization without loss of mass); and finally, the ADS is actuated to assist the EHRS in depressurizing the RV. The containment pressure is limited by the PSS and the reduced break flow due to the EHRS heat removal from the RV.

At the end of the blowdown phase, the RV and CV pressure become equal (pressure equalization) with a CV pressure peak less than 8 bar_g. The break flow stops and the gravity makeup of borated water from the suppression pool becomes available.

The coupled RV/CV system is then depressurised (RV/CV depressurization phase) by the EHRS (steam condensation inside the RV exceeds decay heat boil-off). In this phase the break flow reverses since heat is removed not from the containment, but directly from inside the vessel, and this increases the liquid level in the vessel. As steam from the containment is condensed inside the pressure vessel (RV and CV pressure reduced to less than 2 bar_g within 12 hours), the containment pressure is reduced, and a portion of suppression pool water is pushed out through the vents and assists in flooding the vessel cavity.

The depressurization phase is followed by the long term cooling phase where the RV and CV pressure is slowly reduced as the core decay heat decreases.

During this phase of the accident recovery, gravity makeup of borated water from both suppression pool and RV cavity is available as required. Since decay heat is directly removed from within the vessel and the vessel and containment are thermodynamically coupled, the long term break flow does not correspond to the core decay heat, but it is in fact limited to only the containment heat loss.

- **Steam generator tube rupture** – In IRIS, the steam generator tubes are in compression (the higher pressure primary fluid is outside the tubes) and the steam generators headers and tubes are designed for full external reactor pressure. Thus, tube rupture is much less probable and if it does occur, there is virtually no chance of tube failure propagation. Apart from reducing the probability of the event occurrence, IRIS also provides by design a very effective mitigation to this event.

  Since the steam generators, the feed and steam piping and the isolation valves are all designed for full reactor coolant system pressure, a tube rupture event is rapidly terminated by closure of the faulted SG main steam and feed isolation valves upon detection of the failure. Once the isolation valves are closed, the primary water will simply fill and pressurize the faulted steam generator terminating the leak. Given the limited volume of the steam generators and piping, no makeup to the RV is even required; and since the faulted SG is immediately isolated, the release of radioactivity (primary fluid) to the environment will be minimized.
- **Increase in heat removal from the primary side** – The limited water inventory in the once through steam generator has an important effect on the events in this category. Increases in heat removal due to increased steam flow are eliminated since the steam flow from the once through steam generators cannot exceed feed water flow rate. Also, the consequences of a design basis steam line break event are significantly lessened. Not only is the impact on the containment limited by the reduced discharge of mass/energy, but also no return to power due to the cooldown of the primary system is possible.
- **Decrease in heat removal from the secondary side** – Events in this category (which include loss of offsite power, loss of normal feedwater, turbine trip and feed system piping failure) could potentially have larger consequences in IRIS than in loop type PWRs because of the limited water inventory in the once through steam generators. However, the IRIS design compensates for the limited SG water inventory.

  The limited heat sink provided by the steam generators is in fact more than balanced by the large thermal inertia in the primary system (the IRIS water inventory is more than five times larger than advanced passive PWRs like AP1000 on a coolant mass-per-MWth basis), and by the large steam volume in the IRIS pressurizer (steam volume-to-power ratio is also more than five times that of the AP1000). The reactor trip setpoint is rapidly reached on a low feedwater signal, and the EHRS connected to the steam generators effectively removes sufficient heat to prevent any pressurizer overfill or high pressure relief from the reactor vessel to the containment.

- **Decrease in reactor coolant flow rate** – The IRIS response to a complete loss of flow is comparable to that of the AP600/AP1000, where the coastdown of the reactor coolant pumps is sufficient to maintain core cooling until the control rods are inserted and power is decreased. For the design basis locked rotor event, the IRIS response is improved over other PWRs by the increased number of reactor coolant pumps, which reduces the relative importance of a loss of a single pump flow. This design choice allows IRIS to prevent fuel damage (i.e., no departure from nucleate boiling) following a postulated locked rotor event even without a reactor trip. Of course a shaft break accident cannot occur, because spool pumps do not have shafts.
- **Spectrum of postulated rod ejection accidents** – Locating the CRDMs internally to the reactor vessel eliminates by design the rod ejection accident since there is no significant driving differential pressure over the driveline.
- **Increase in reactor coolant inventory** – This category of events is eliminated in IRIS since IRIS does not utilize high-pressure coolant injection following a LOCA. The inadvertent actuation of the small emergency boration tanks can be accommodated by the large pressurizer volume with no overpressure or overfill of the RV.

6.3. Possible practical consequences of the safety-by-design™

The superb safety characteristics of the IRIS design offer the possibility of pursuing a relaxation in current licensing requirements, which would have a very positive effect both in terms of overall plant economics and of public acceptance.

The overall approach to safety in IRIS may be represented by the following three-tier approach:

1. The first tier is the already discussed safety-by-design™, which aims to eliminate by design the possibility for an accident to occur, rather than dealing with its consequences. By eliminating some accidents, the corresponding safety systems (passive or active) become unnecessary as well.
2. The second tier is provided by simplified passive safety systems, which protect against the still remaining accidents and mitigate their consequences.
3. The third tier is provided by active systems, which are not required to perform safety functions and are not considered in deterministic safety analyses, but may contribute to reducing the core damage frequency (CDF).

The third tier has been addressed within the PRA framework. In fact, PRA was initiated early in the IRIS design, and was used iteratively to guide and improve the design safety-wise. The PRA has suggested certain modifications to the layout that were implemented, resulting in a reduction of predicted CDF. After these modifications, the preliminary PRA level 1 analysis [9] estimated the CDF due to internal events to be 1.2×10^{-8}. This value is more than one order of magnitude less that in advanced LWRs, and it is dominated by the vessel rupture with its “given” CDF of 1×10^{-8}. Achieving such low CDF value was possible due to the inherent amenability of the integral configuration to improved safety, combined with the safety-by-design™ and PRA-guided design improvements.

The defence-in-depth provided by the safety-by-design™ as the first step, which results in the elimination or lessening of Class IV events (only one left out of eight typically considered, as shown in Table 4), combined with the low CDF (more than one order of magnitude improvement) and risk-informed licensing could allow IRIS to attain ambitious licensing objectives, such as the licensing with no requirements for off-site emergency response planning (i.e., emergency planning zone equal to site exclusion zone). This objective, also declared by the AGENCY as one of the top-level goals for advanced reactors, would have a significant positive socio-economic impact.

Economically, the utility / plant operator is not required to plan for emergency evacuation, allowing a larger choice of sites, and avoiding the expenses of physically preparing the site and conducting planning for emergency response.

Public acceptance will be greatly improved, because essentially IRIS will be declared to be no different from other power producing plants, by removing the "red flag" associated with nuclear plants.

Licensing without the off-site emergency response requires elaboration of a new licensing framework, which the IRIS project will explore with the U.S. NRC during the current pre-application process, as well as by keeping close contacts with the AGENCY.

7. PROCEEDING TOWARDS COMMERCIALIZATION

IRIS has been recognized internationally as an attractive advanced LWR with significant market potential, as demonstrated by the Consortium membership. It figures prominently within the International Near-Term Deployment (INTD) group established by the Generation IV International Forum (GIF). From the utility side, it was included in the Early Site Permit (ESP) program, pursued by three power utilities (Dominion, Entergy, and Exelon) with the support of U.S. DOE. In this program, utilities can "pre-qualify" their existing sites for construction of a new nuclear power plant. However, rather than licensing the site for a specific design, the utility develops an envelope of site requirements that encompass design characteristics of all candidate designs.

In response to such utilities' request, IRIS has developed two alternative site layouts and corresponding site requirements:

(a) Multiple single-unit site layout;
(b) Multiple twin-unit site layout.

In the first option, illustrated in Fig. 7, shared systems and structures are minimized. Units are constructed in a "slide-along" manner, with first unit put into operation while subsequent units are under construction. Such arrangement minimizes construction time and provides generating capacity (and revenue) as soon as possible. It also maximizes workforce efficiency and significantly shortens construction time of subsequent units.

In the second option, shared systems and structures are maximized (including fuel handling and spent fuel pool, support systems in auxiliary building). Twin-units share control rooms, but have separate safety and protection systems. Twin-units are also constructed in "slide-along" manner, with the same advantage as for the first option, plus maximization of shared equipment and workforce, but it requires adding generating capacity in 670 MWe increments.

For the U.S. ESP program, the requirement was to provide at least 1000 MWe per site, and the optimum arrangement was found to be a two twin-unit site with 1340 MWe total installed capacity. However, the 1000 MWe requirement may not be appropriate in case of smaller grids and/or countries that don't need (or cannot afford) such a large addition at once. Instead, multiple single-unit site layout allows starting with a single unit, and adding 335 MWe at a time, as the need warrants. This reduces investment and improves cash-flow, as the first unit starts to generate revenues three years after the initial construction.

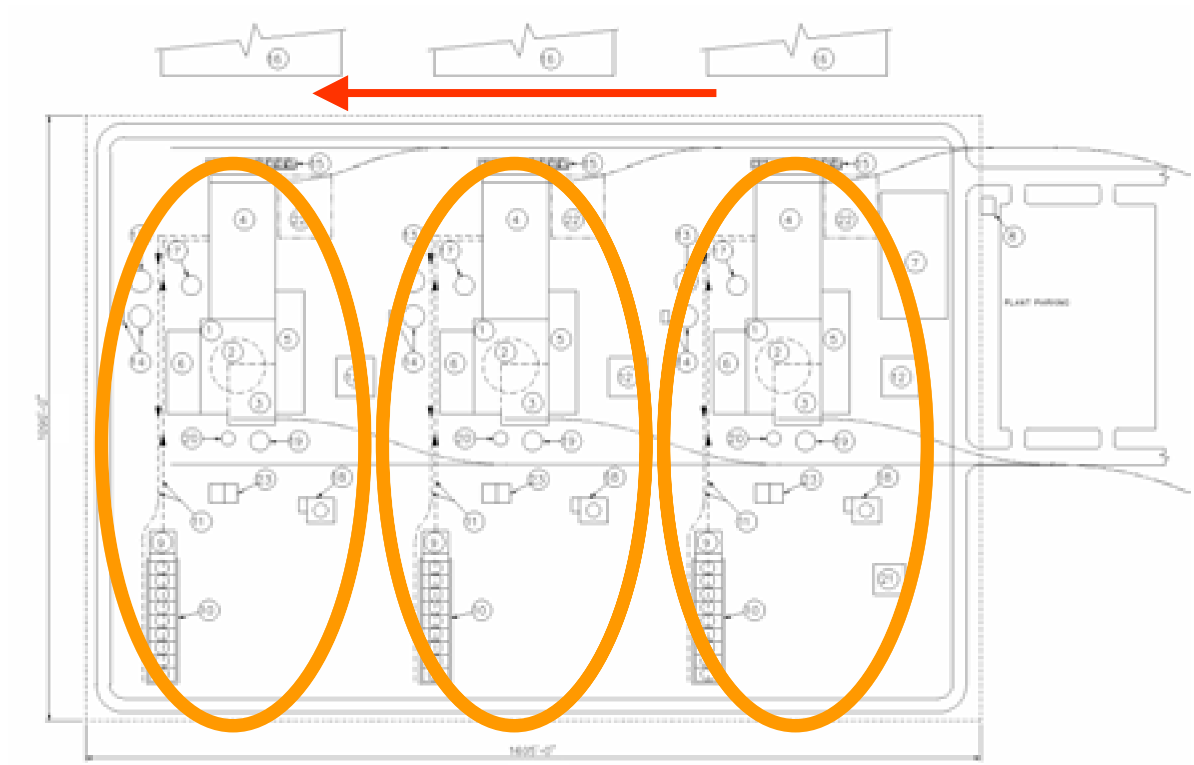

FIG. 7. IRIS multiple single-unit site layout.

It should be acknowledged that IRIS cannot rely on the traditional economy of scale. Instead, it offers economy of identical multiples, lower financing requirements, and faster response to market needs. The first of these advantages, economy of identical multiples, is not limited to having several modules being built at the same time at the same place. A similar benefit may be achieved through the IRIS Consortium, if a construction of a single unit each takes place at several sites at about the same time, e.g., at several countries with smaller grids (to match limited grid needs), and/or at several sites throughout a larger region to fit regional requirements. This also allows an accelerated learning curve through accumulating experience on the same design under different conditions. To fit specific needs of certain market segments, IRIS plant designs modified for district heating or desalination are being developed.

IRIS economics relies on short construction time, optimised maintenance, and high reliability. A preliminary top-down economics assessment has been performed [16] indicating that IRIS will be competitive in all geographic regions. If additionally licensing without the off-site emergency planning may be achieved, it would not only provide a significant financial benefit (e.g., no need to build evacuation routes; siting closer to consumers thus reducing transmission costs), but it would also improve public acceptance.

8. CONCLUSIONS

An overview of the status of the IRIS design has been provided, with particular emphasis on the integral layout of the reactor coolant system and on the innovative IRIS approach to safety.

The integral layout offers very significant advantages in terms of performance, simplicity, and compactness. It has been demonstrated that it has an extremely positive impact on the overall reactor safety response to postulated accidents. It is also expected to have a positive economic impact and work has been initiated for its verification.

Because of the safety-by-design™ approach, the number and complexity of the safety systems and required operator actions are minimized in IRIS. The net result is a design with significantly reduced complexity, improved operability, and extensive plant simplifications. Moreover, the safety-by-design™ combined with very low predicted CDF, has the potential to enable licensing with no need for off-site emergency response, with additional significant positive economics impact.

Due to this medium size, IRIS is particularly well suited for countries or regions with a small or medium electricity grid. It is ideal for utilities that cannot assume large capital investment or risk that is associated with large power plants, and that require a gradual increase of generating capacity. A preliminary top-down economic analysis indicates that IRIS is expected to be competitive in all geographic regions, due to its optimised maintenance, simple configuration, short construction period, and high availability. Co-generation design options for desalination and district heating are also being developed to address needs of specific market segments. Overall, IRIS has a large potential in the worldwide market.

LIST OF ACRONYMS AND ABBREVIATIONS

ADS	Automatic Depressurization System
ATWS	Anticipated Transient without Scram
CRDM	Control Rod Drive Mechanism
CV	Containment Vessel
DID	Defence in Depth
DOE	Department of Energy
DVI	Direct Vessel Injection
EBT	Emergency Boration Tank
EHRS	Emergency Heat Removal System
ESP	Early Site Permit
IRIS	International Reactor Innovative and Secure
LOCA	Loss of Coolant Accident
MIT	Massachusetts Institute of Technology
NERI	Nuclear Energy Research Initiative
NRC	Nuclear Regulatory Commission
O&M	Operation and Maintenance
PBMR	Pebble Bed Modular Reactor
PRA	Probabilistic Risk Assessment
PSS	Pressure Suppression System
PWR	Pressurized Water Reactor
RCCA	Rod Control Cluster Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Reactor Vessel
RWST	Refuelling Water Storage Tank
SG	Steam Generator

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SCOR: SIMPLE COMPACT REACTOR — AN INNOVATIVE MEDIUM SIZED PWR

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Abstract. A preliminary study for the development of innovative medium sized pressurized water reactor design SCOR (Simple Compact Reactor) is undertaken to meet the established requirements for future electricity generating reactors, such as low power cost, high level of safety, effective fuel cycle strategy, and the provisions of the European Utility Requirements (EUR). This conceptual design makes account of potential markets for medium sized reactors, reflects the intention to have a compact nuclear steam supply system, provides for the possible use of innovative fuels and for simplification of the safety demonstration through elimination of some accidents by design, incorporates many passive safety systems, and ensures low cost of generated power through optimised operation, fuel cycle and investments. The concept appears as a compact reactor with integrated pumps, pressurizer and control rods. A single steam generator is located above the reactor vessel within the reactor closure head. The decay heat is passively removed by dedicated integrated heat exchangers.

1. INTRODUCTION

For future reactors the utilities require an improvement of safety, competitiveness and better integration of the present-day fuel cycle constraints (reference is made to European Utility Requirements). For this purpose new concepts of nuclear systems are proposed by the nuclear industry.

In the framework of innovative reactor studies conducted in the 1990ies, the CEA has evaluated pressurized water reactors (AP 600, SIR, PIUS, low pressure PWRs), safety systems, and several types of PWR cores.

The demand for future reactors along with the expertise acquired by the CEA resulted in a proposal of a 600 MWe medium-sized pressurized water reactor design entitled SCOR (Simple Compact Reactor).

2. SELECTION OF REACTOR ARCHITECTURE AND UNIT POWER

The approach in selection of design parameters was as follows:

- Compact reactor was selected in order to reduce the overnight investment costs, particularly as related to the building architecture;
- Power level was selected to be compatible with the above option, which also matches the potential market for electricity generating reactors [1,2];
- To a degree possible, accidental conditions that require costly management systems were eliminated at the design stage. Maximum use was made of the knowledge acquired from present-day reactors and existing components;
- The reactor core was designed to accommodate innovative fuel;
- Such approach resulted in the selection of a 600 MWe integrated design pressurized water reactor with the following features (see Fig. 1);
- Core with a power density lower than that of current PWRs to facilitate the use of innovative fuels;
- A single steam generator acting as the reactor vessel head;
- Integrated control rod drives;
- An integrated pressurizer;
- Integrated pumps.

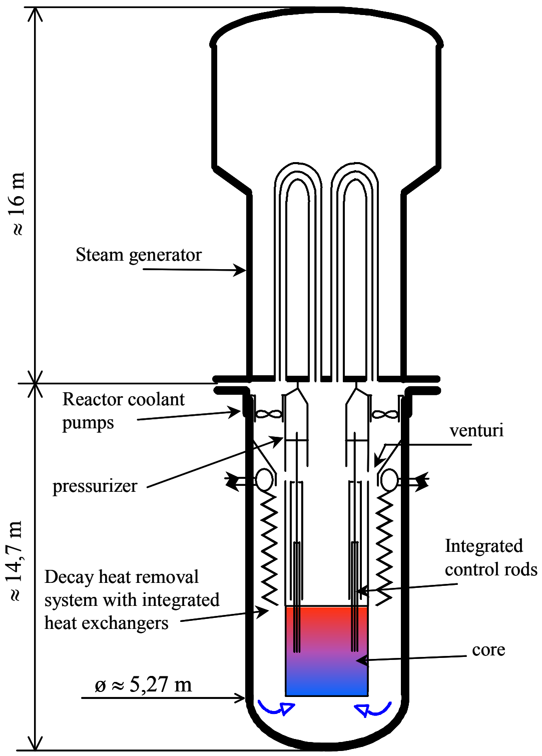

FIG. 1. Diagram of the SCOR design.

3. OPERATING PARAMETERS

According to the results of low-pressure PWR studies, an operating pressure of about 85 bars may be appropriate for the SCOR [3].

Relative to the operating point of current PWRs, low pressure and low temperature may result in:

- A considerable reduction of the thickness of pressurized components (reactor vessel, steam generator (SG), etc.);
- A potential increase of fuel burn-up (less cladding corrosion);
- Simplification of safety systems.

The pressure at the turbine inlet in secondary circuit is lower than that of a standard PWR, which results in a somewhat lower thermodynamic efficiency. As a first approximation, the net efficiency was estimated to remain above 30% for the 30 bar pressure. The thermal power of the core is set at 2000 MW.

4. CORE DESIGN

The core is identical to that of a French 900 MWe PWR, and as such consists of 157 square assemblies of 17×17 fuel rods each. In this, the specific power is lower by 28% relative. The advantages of lower power density are as follows:

- Longer cycle duration obtained through lower power density ensures an increased availability;
- Elimination of soluble boron and implementation of innovative fuels are facilitated by taking advantage of higher critical heat flux margins obtained through the combined application of lower operating parameters and lower power density;

- The core power and power density are a priori compatible with an in-vessel corium retention strategy based on reactor vessel pit flooding and, therefore, an external core catcher is not required.

5. REACTIVITY CONTROL

5.1. Integrated control rod drives

The use of integrated drives is necessary as standard control rod drives (CRDs) are incompatible with a SG placed above the reactor vessel. Internal CRDs also eliminate the risk of rod ejection, e.g. due to rupture of a CRD nozzle on the reactor vessel head, and therefore remove the constraints associated with reactivity insertion accident in the determination of maximum discharge burn-up.

The hydraulic drives that were developed for the Boiling Water Reactor (BWR) [4] consist of a hollow piston and a mobile cylinder (see Fig. 2). The piston is attached to the core support plate, and the cylinder is connected to the absorbing elements. Along their entire height, the piston and lower part of the cylinder are uniformly grooved. The cylinder position is maintained by introducing a given primary fluid flow rate into the interior of the piston. The cylinder is made to rise or lower along a distance equal to the groove pitch by temporarily increasing or decreasing the primary fluid flow rate.

The proposed hydraulic drive for the SCOR (see Fig. 3) is inspired by the above principle. It consists of a hydraulic drive incorporated into the upper plenum of the reactor vessel. The cylinder is fixed to the upper plate of the cluster guide and is grooved along its entire height. The upper portion of the mobile piston is also grooved.

The displacement piston acts as a rod drive and is used to move the absorbing rod clusters, which are identical in design to those in the standard PWRs.

5.2. Soluble boron free core

The selection of a soluble boron-free core is based on the studies conducted at the CEA [5]. The main design features allowing for this type of core in SCOR reactor are:

- 10% increase in the moderation ratio due to an increase in the moderator density;
- The accommodation of local power peaks slightly more intense than a standard PWR due to the low power density;
- The possibility to use one control cluster for two assemblies or even one per assembly due to the compact integrated hydraulic drives;
- The elimination of a potential blockage of several CRDs in the case of a large primary break, due to the integrated design;
- Elimination of local power excursions due to the risk of rod ejection accident, by the use of integrated CRDs;
- Less strict reactivity control requirements due to low power density (reduced power and Doppler reactivity effects) and the lower operating point parameters (reduction of the absolute value of moderator coefficient).

The selection of boron-free core leads to a considerable simplification of the auxiliary systems related to boron management, which in turns leads to a significant reduction in investment and maintenance costs. Reactivity control in accidents may be backed-up by the injection of borated water.

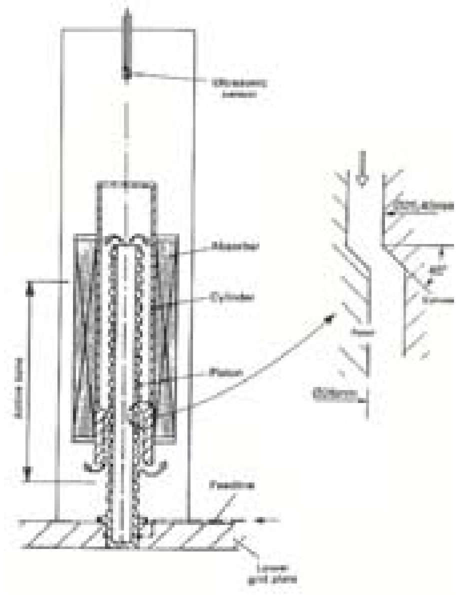

FIG. 2. Scheme of the BWR integrated rod drive.

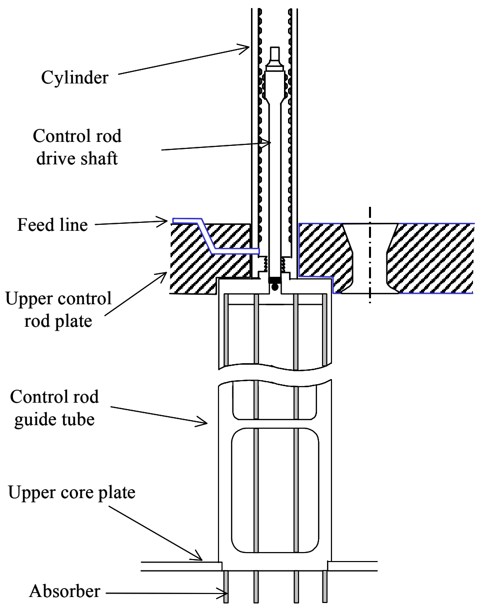

FIG. 3. Hydraulic drive being customized to the SCOR design.

6. STEAM GENERATOR

There is only a single U-shaped boiler-type steam generator. As in the case of some propulsion reactors, the SG is placed above the core. In contrast to standard SGs the present generator has an axial symmetry. The hot leg is placed in the centre and encircled by the cold leg.

7. HEAT EXCHANGER-PUMP MODULE

Large annular space located between the core barrel shell and the reactor vessel accommodates the heat exchanger-pump modules. Each module (see Fig. 4) consists of a primary fluid circulation pump and a heat exchanger used to remove residual heat.

The pump, placed in the upper part, is supplied with water from the steam generator. The submerged coil-type motor is located downstream of the wheel. The water flows around the motor, is accelerated by a venturi, passes into a diffuser and then goes through the heat exchanger tube bundles.

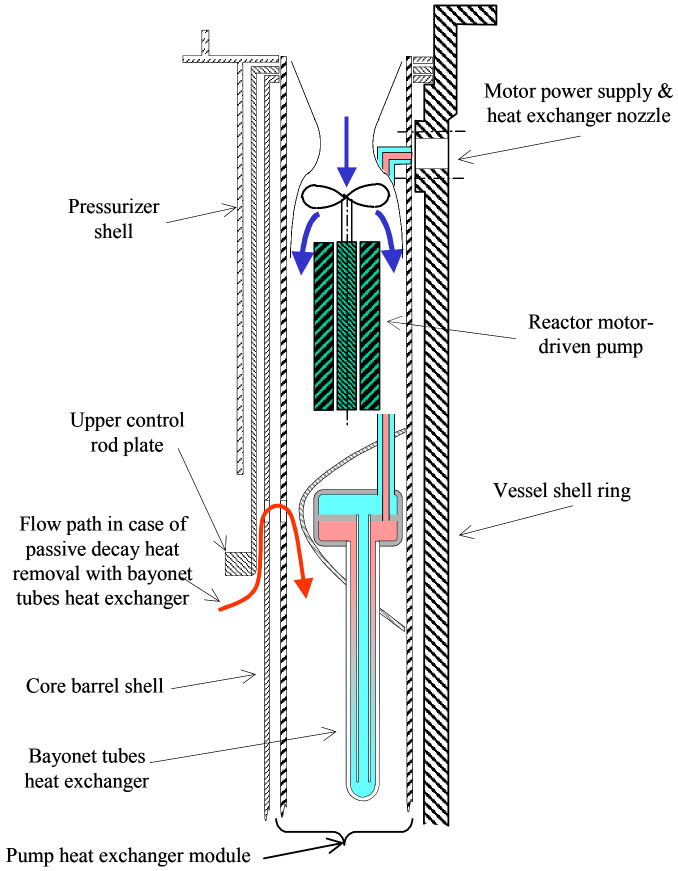

FIG. 4. Heat exchanger-pump module.

The heat exchanger consists of bayonet tubes with their outside surface being wetted by the primary fluid descending within the module. The secondary water flows first in the internal tube and then upwards through the annular space bounded by the two tubes. The water box is located in a dead zone behind the venturi. Heat exchanger of this type does not require a water box at its exit. This reduces the primary pressure drop and allows free expansion of the tubes. The thermal loadings are reduced, which leads to an increased mechanical resistance and an enhanced reliability SG bypass, installed within the venturi between the core exit and the cold leg, allows natural convection of the primary fluid during the pump shutdown. During normal operation, high flow velocity at the venturi throat decreases the pressure locally. The cross section area of the venturi throat is designed to balance pressure between the hot leg (core exit) and the cold leg (heat exchanger-pump module), in order to have no bypass flow.

The module can be removed from the reactor vessel once the steam generator has been removed. The pump power supply and the heat exchanger secondary feed-lines are set in place via a removable opening in the upper part of the reactor vessel.

8. PRESSURIZER

The volume of the pressurizer is approximately 2 times smaller than that of a standard 900 MWe PWR, due to the lower specific core power and low operating pressure and temperature (leading to lower variations in the coolant density). The pressurizer is located in the reactor vessel just below the SG. It has an annular shape resembling the inverted U (Fig. 5). The electric heaters are placed in a small volume tank outside the reactor vessel and act as a steam source. The cold water is tapped off just downstream of the pumps and the two-phase mixture is re-injected at the top of the pressurizer.

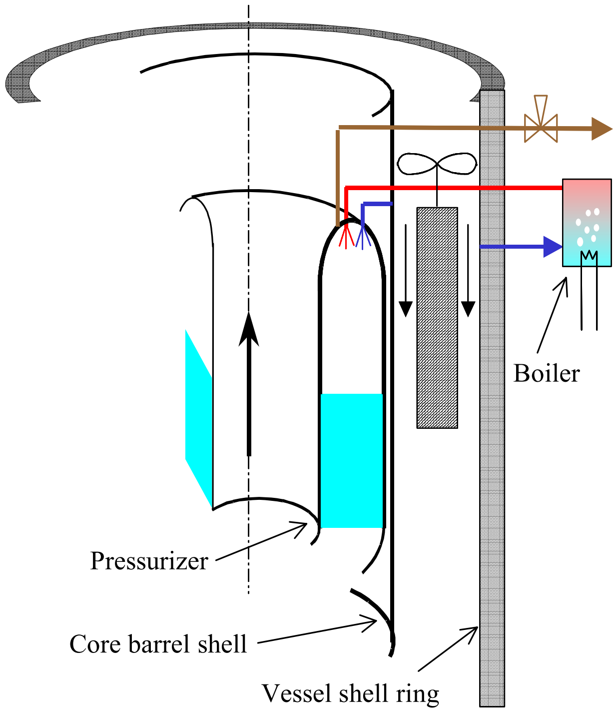

FIG. 5. Pressurizer scheme.

9. SAFETY SYSTEMS

Safety systems must satisfy the recommendations for future reactors. The design of safety systems is in progress.

9.1. Decay heat removal

Since the reactor has only one steam generator, the decay heat removal systems should be provided in both primary and secondary circuits in order to ensure the diversity of means.

9.1.1.Decay heat removal in the secondary system

In case of an accident, the heat removal device should not release steam during a steam generator tube rupture (SGTR). The heat sink could be provided by immerged heat exchangers. Two possible arrangements are compatible with passive operating mode:

- either a pool located above the steam generator with heat transfer control via a thermal valve [6, 7], see Fig. 6; or
- a pool located at ground level with heat transfer control via a steam injector [8] see Fig. 7.

9.1.2. Decay heat removal in the primary system

The primary system is cooled by heat exchangers located within the heat exchanger-pump modules. The venturi with a bypass provides natural convection in primary system irrespective of steam generator and pump states.

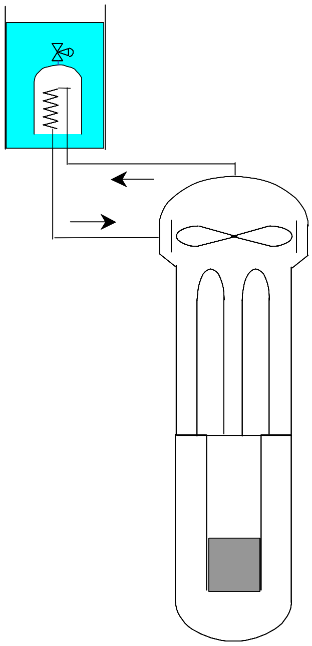

FIG. 6. Secondary system decay heat removal with a condensation system and a thermal valve.

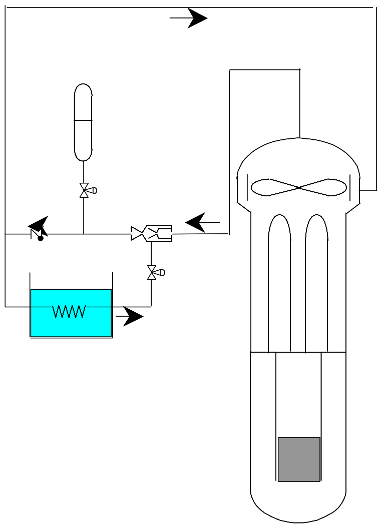

FIG. 7. Secondary system decay heat removal with steam injector.

Such system is able to cool the primary system down to the cold shutdown state and replace the normal reactor heat removal system. Heat exchangers submerged in a pool constitute the heat sink for half of the modules. Other modules are supposed to employ cooling tower systems (Fig. 8).

9.2. Safety injection system

Since large LOCAs in SCOR are eliminated by design and since the primary system thermal inertia is higher than that of conventional loop-type PWRs, the safety injection system could be based on devices operating with low flow rate.

Under the selected low-pressure option, safety injection system of only one type with an operating pressure of about 25 bar (instead of 110, 40 and 20 bar in a standard PWR) can be envisaged.

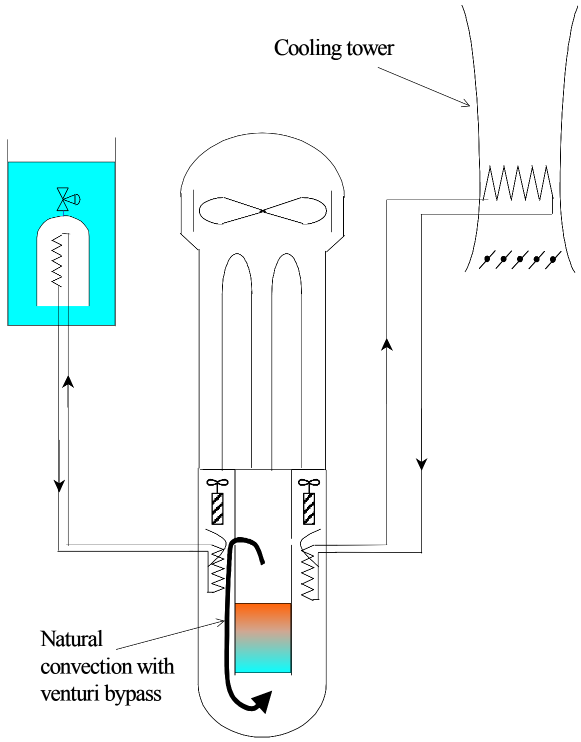

FIG. 8. Decay heat removal in the primary circuit.

9.3. Primary system pressure control

Studies conducted for management of accident scenarios have shown that safety systems using a heat exchanger in the primary system are very efficient [9]. Supposing the failure of all engineered safety systems in primary circuit except the passive system, it is possible to manage transients without using a safety depressurisation system in primary circuit, while avoiding core meltdown under high pressure.

10. CONTAINMENT

SCOR is a compact reactor in many ways similar to BWRs and the SIR concept [10]. The containment is assumed consist of a pressure suppression compartment. The specific feature of the SCOR design is that it includes two physically separated areas: one located below the mating surface that couples the reactor and the SG and another one located above this surface. The upper area contains the reactor vessel with all its small diameter pipe connections. The lower area is in contact with only the secondary system when the reactor is in operation, and with the primary system during maintenance operations.

According to the studies in progress, a containment design with two compartments could be achievable: a Reactor Vessel Compartment that houses the reactor pressure vessel, its connections and the support structure, and a Steam Generator Compartment that houses the steam generator and the refuelling cavity (see Fig. 9).

11. SEVERE ACCIDENTS

Compared to a standard PWR, the safety of SCOR is facilitated by the elimination of some initiating events at the design stage. Among them are large breaks in the primary circuit and reactivity insertion accident as initiated by control rod ejection.

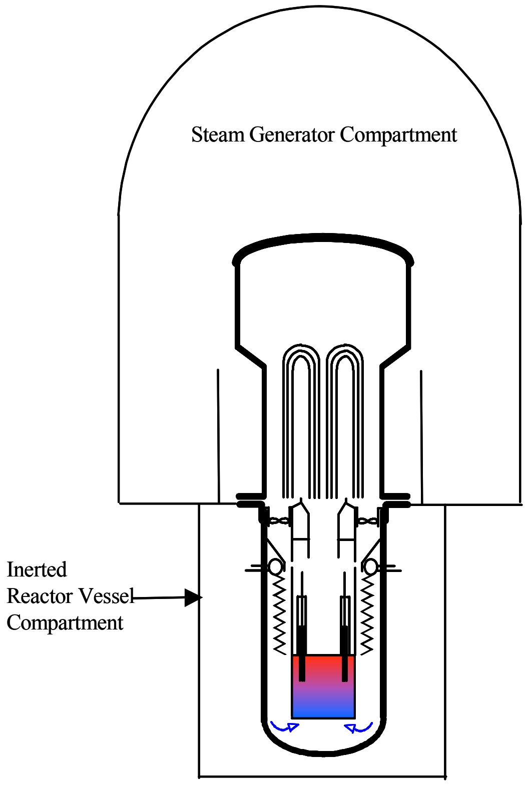

FIG. 9. Scheme of the containment.

However, it is assumed that the hypothetical case of core meltdown should be manageable through implementation of the following measures:

- Core meltdown: corium cooling by reactor vessel pit flooding;
- Hydrogen risk: the Reactor Vessel Compartment atmosphere is kept inert to prevent hydrogen combustion.

12. PRE-DIMENSIONING OF THE SCOR DESIGN

12.1. Pre-dimensioning tool

The initial dimensioning of SCOR design was performed using the COPERNIC tool developed at the CEA. COPERNIC (COde de Pré-dimensionnement et d'Evaluation des Réacteurs iNnovants par la méthode d'Ingénieries Concurrentes, *pre-dimensioning and evaluation code for innovative reactors by competitive engineering methods*) is a tool that provides assistance in the preliminary design of innovative reactors. Being thoroughly based on the published data and incorporating multiple feedbacks from its practical application, this code reflects the state-of-the-art in simulation of nuclear steam supply systems. Specifically, this code offers an extensive database linked to an expert nuclear reactor study system. It provides for a quick design evaluation and defines consistent data sets for the subsequent in-depth studies.

12.2. Main characteristics of SCOR

The main characteristics of SCOR are summarized in Table 1 below, with PWR-900 data being given as reference.

TABLE 1. DESIGN DATA AND MAIN CHARACTERISTICS OF SCOR

Reactor design		PWR	SCOR
Thermal power	(MW)	2771	2000
Net electric power	(MW)	915	630
Net efficiency	(%)	33.0	31.5
Primary system			
Primary pressure	(bar)	155.0	88.0
Core inlet temperature	(°C)	286.5	246.4
Core outlet temperature	(°C)	324.9	285.9
SG inlet temperature	(°C)	322.4	284.3
Primary coolant flow	(kg/s)	13498	10465
Core			
Active core height	(m)	3.66	3.66
Number of 17×17 assemblies		157	157
Power density:	(kW/l)	104.8	75.3
Reactor vessel			
Outside diameter (OD)	(m)	4.448	5.265
Annular space width	(m)	0.270	0.744
Height	(m)	12.03	14.67
Empty vessel weight	(t)	320	277
Primary water volume off-PRZ* (m^3)		240	273
Primary inertia	(m^3/MW)	0.086	0.136
Secondary system			
Number of SGs		3	1
Steam pressure	(bar)	58.0	32.0
Steam temperature	(°C)	273.3	237.4
Number of tubes per SG		3 330	11 000
Tube length	(m)	20.22	16.26
Heat transfer area per SG	(m^2)	4701	10707
Tube OD	(mm)	22.22	19.05
Tube thickness	(mm)	1.27	0.80
SG height	(m)	20.65	16.06
Weight of a single SG	(t)	314	569
Pressurizer			
Total volume	(m^3)	40.0	21.3
Full power liquid volume	(m^3)	24.5	14.0
Primary pumps			
Water head	(m)	87.8	44.4
Number of pumps		3	16
Electric power per pump	(kW)	5301	447
Heat exchanger-pump module			
Number of modules			16
HX:Outer tuve OD			30.0
HX: Outer tube thickness	(mm)		1.50
HX: Internal tube OD	(mm)		19.0
HX: Internal tube thickness	(mm)		1.50
Pitch	(mm)		42.0
Number of tubes per heat exchanger			153
Bundle height	(m)		4.66
Economic indicators			
NSSS** mass to reactor output ratio. (t/MWe)		2.02	1.92

* PRZ is for pressurizer

** NSSS is for **N**uclear **S**team Supply **S**ystem

12.3. Decay heat removal system performance

The COPERNIC tool was used to evaluate the decay heat removal system, assuming natural convection in the primary system. The primary circuit consists of the core, riser-module path, section downstream from the venturi, and heat exchangers of the heat exchanger-pump module. The pumps and SG were assumed to be unavailable.

To remove 4% of the nominal power, which corresponds to a few minutes after reactor shutdown, the hot and cold temperatures are 280 and 248°C on the primary side, and 177 and 120°C on the secondary side (see Fig. 10).

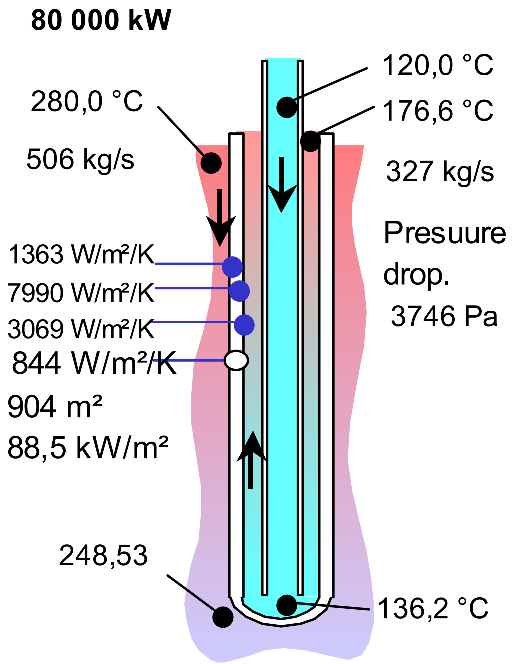

FIG. 10. Performance data of the decay heat removal system.

Small pressure drop (less than 4,000 Pa) on the secondary side of the heat exchanger is compatible with a system based on natural convection. Secondary system temperatures are compatible with a heat sink made of either the immerged heat exchangers or a cooling tower. System performance data are therefore compatible with an entirely passive decay heat removal.

13. ECONOMICS

It is widely accepted that the investment cost decreases with the reactor power according to the 'power law':

$$C(P_1)= C(P_0)\times(P_1/P_0)^n,$$

where n is between 0.4 and 0.7, and C(Po) is a design-dependent coefficient.

The specific mass of NSSS (reactor vessel and its internal equipment, SG, pumps, pressurizer and main primary pipelines) makes it possible to illustrate this law, even if the investment cost is not directly proportional to the specific mass. This power law can have discontinuities if the designs include threshold effects.

A quantitative illustration of this law and the threshold effects is carried out through a brief analysis of different reactor designs [10], (See Fig. 11).

The specific mass of the primary system of an integrated reactor that operates with natural convection is near 8 to 10 t/MWe and the technological limit appears to be around 100 MWe. In the case of an integrated design operating under forced convection (like the SIR design), the specific mass is around 4 to 5 t/MWe and the limit is around 300 to 400 MWe. For higher power an integrated design is no longer possible due to unmanageable reactor vessel sizing; hence loop-type design becomes the option. For such design it is possible to demonstrate the presence of threshold effects, such as the in-vessel corium retention by flooding of the reactor vessel pit, which should be possible if the core power is less than 2,000 MWth (AP 600 case). This threshold effect is difficult to estimate, because it implies a simplification of the safety systems, and does not intervene in the specific mass balance of the primary system. French PWRs with a power higher than or equal to 900 MWe have a specific mass slightly above 2 t/MWe.

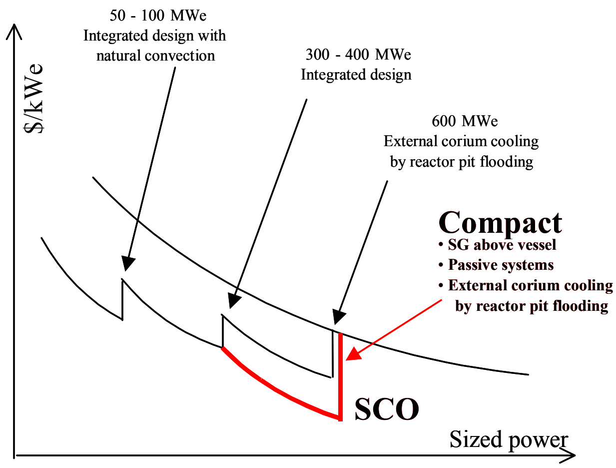

FIG. 11: Visualization of threshold effects in the cost of kWe.

SCOR is a design in progress, and at its present stage of development it is not possible to estimate the investment cost. However, in selecting the design options, we opted for a high-powered reactor in order to reduce the investment cost (as dictated by the power law), while simplifying the auxiliary systems by combining a maximum possible number of threshold effects. The qualitative arguments given below suggest that a competitive overall cost with the overnight investment cost reaching 1,000 $/kWe could be targeted for SCOR:

- Specific mass of the primary system is slightly less than that of a high powered loop-type PWR;
- Elimination of soluble boron leads to the elimination of a number of systems, and furthermore allows a reduction in the dose rates associated with the auxiliary systems and tritium waste, leading to a reduction in operation and maintenance costs;
- Increase in the discharge fuel burn-up is facilitated by the decreased operating temperatures (less corrosion);
- Availability could be increased due to the higher discharge burn-up and lower specific power, which should allow cycle durations of at least two years between refuellings;
- Reactor vessel life could be prolonged since the neutron fluence upon reactor vessel is reduced through the increase in annular space width;

- The integrated design and low operating pressure reduce the number of safety systems and their maintenance costs;
- Compact pressure suppression containment should be less costly than the containment of standard PWRs;
- Absence of an external core catcher and easier management of the hydrogen risk simplify the severe accident management systems;
- Simplifications in the SCOR design reduce the construction time.

14. CONCLUSION

A preliminary study for the development of innovative medium sized pressurized water reactor design SCOR (Simple Compact Reactor) was undertaken to meet the established requirements for future electricity generating reactors, such as low power cost, high level of safety, effective fuel cycle strategy, etc. The general architecture and design options for SCOR were selected through design evaluation studies performed within the CEA's innovative reactor programme. The SCOR design uses a single steam generator located above the reactor vessel and acting as reactor vessel head. The primary pumps, the pressurizer and the control rod drives are integrated into the reactor vessel.

The main features of this design are as follows:

- Medium size to suit the potential markets for future electricity generating reactors;
- An option to accommodate cores with innovative types of fuel;
- Strong reliance on passive safety systems;
- Simplified safety demonstration through the elimination of some accident conditions by design.

The simplicity and compactness of the design, as well as simplification of the auxiliary systems and reduction of their number make it possible to project that an overnight investment cost for the SCOR could be as low as 1,000 $/KWe.

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ANNEX 4

NUCLEAR DESALINATION TECHNOLOGY DEVELOPMENT USING SMART

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Abstract. Nuclear desalination technology development using the SMART reactor, which is an integral type pressurized water-cooled reactor with a rated thermal power of 330 MW developed by the Korean Atomic Energy Research Institute (KAERI), is actively in progress. The SMART reactor is coupled with the Multi-Effect Distillation Thermal Vapour Compression (MED-TVC) process plant developed by the Doosan Heavy Industries & Construction Co. Ltd. to produce potable water by seawater desalination. Potable water and electricity from the SMART desalination plant can be supplied to an area with the population of approximately 100,000 or to an industrial complex. A one-fifth scale pilot plant is being constructed to verify safety and performance of the SMART reactor and to demonstrate relevance of the technologies applied for coupling of the nuclear and seawater desalination plant. An international cooperation program is in progress involving the Republic of Korea, Indonesia and the IAEA to study the feasibility of constructing a SMART nuclear desalination plant at the Madura Island in Indonesia.

1. SMART DEVELOPMENT PROGRAMME

The SMART development programme was launched in November 1996. Before this program started, there was an R&D project that was focused on the investigation of technologies for new and innovative components and included various studies on the concepts of advanced nuclear reactor systems. Based on the results of this previous research and development, SMART was decided upon as an integral type PWR to produce a rated thermal power of 330 MW. The programme of a conceptual study was then extended to a conceptual design programme that started from July 1997 as one of the national medium- and long-term nuclear R&D programs. Comprehensive efforts were made to ensure safety enhancement for the SMART concept by implementing passive safety design systems and intrinsic safety features. Along with the conceptual design, efforts were made for the development of various technologies including design methodology, computer codes, fundamental experiments and tests, etc. The conceptual design was completed in March 1999. Preliminary safety analysis for the selected limiting design basis events was carried out to assure a high safety level of the SMART system. Based on these results, it was shown that the SMART system is properly responding to the design basis events by meeting the safety criteria. Successful completion of the conceptual design for SMART was followed by a three-year basic design phase, which began in April 1999. The basic design of SMART was completed in March 2002 [1,2,3].

The six-years long project of the SMART pilot plant development was launched in July 2002. The objectives of this project are to verify the integral performance of the SMART system and nuclear desalination technologies and to confirm their commercial viability through the construction and operation of a 1/5 scale pilot plant, SMART-P. The SMART-P project is carried out by consortium of governmental organizations and domestic nuclear industries, with strong support from the nuclear community

2. SMART, AN INTEGRAL REACTOR

An innovative approach based on broad implementation of the intrinsic safety features and passive safety technology was adopted in the design of SMART reactor. All primary components, such as core, steam generators, main coolant pumps, and pressurizer are integrated into a single pressurized vessel with no piping used for the connection of these components

Figure 1 shows structural configuration of SMART reactor. Four main coolant pumps are installed vertically at the top of the reactor pressure vessel. The reactor coolant flows upward through the core and enters the shell side of a steam generator from its top. Steam generators are located above the core in a circumferential space between the core support barrel and the reactor pressure vessel. Large volume in the upper part of the reactor pressure vessel (RPV) is used as a self-pressurizer. Such integral arrangement of major primary components within a single RPV is the most notable difference between the design concept of SMART and that of the conventional loop type reactors.

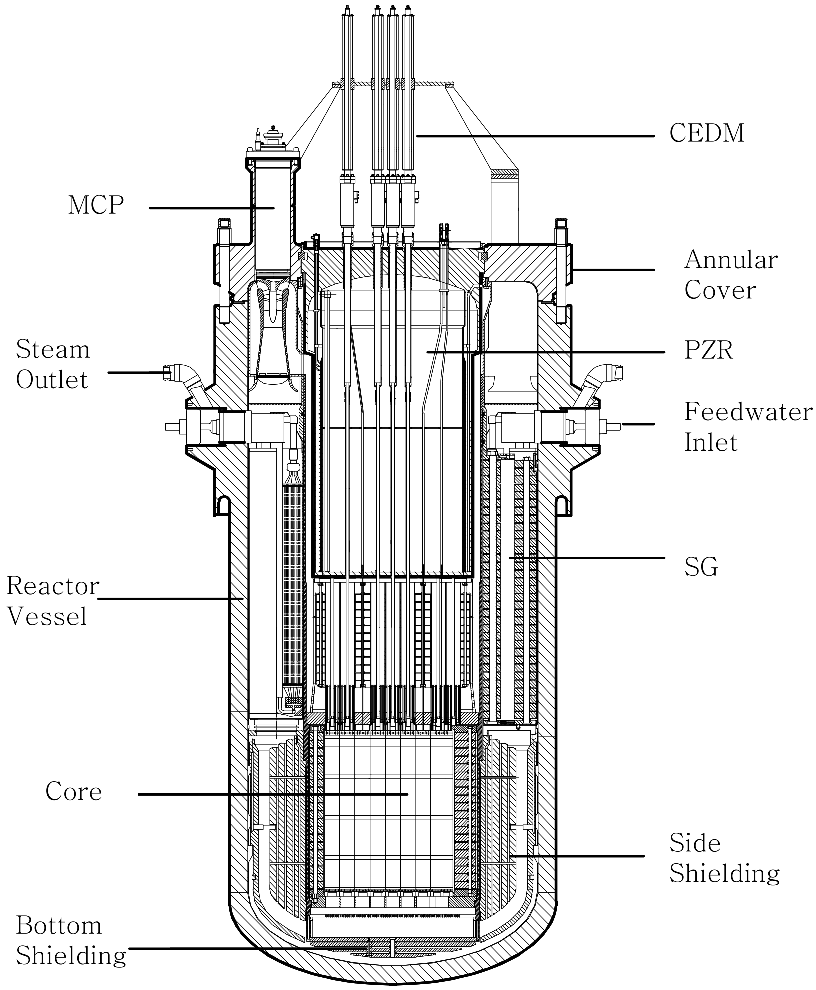

FIG. 1. Structural arrangement of the SMART RPV.

An integral arrangement of the primary reactor systems within a single pressure vessel eliminates large-sized pipe connections between primary components. The adoption of simplified passive systems results in the net reduction of a number of safety systems, and also drastically reduces the number of valves, pumps, wirings, cables, pipes, etc. Elimination of soluble boron system is one of the more important design features largely contributing to the overall system simplification by allowing the elimination of the associated systems and components required for boric acid processing and chemical volume control. This feature is also to minimize the generation liquid radioactive waste and therefore to simplify the associated processing systems.

A simplified modular design approach is applied to all SMART primary components. The optimized and modularized small-sized components allow for easy factory fabrication and direct installation at the site, which leads to the shortening of an overall construction time and schedule. These features are to ensure the construction period of less than three years from the first concrete to fuel load. The compact and integral primary system also eliminates the complexity and extra components associated with conventional loop-type reactors.

SMART is designed for a lifetime of 60 years and has a 3-year cycle with a single or one-and-a-half-batch refuelling scheme. The neutron fluence upon reactor vessel is greatly reduced by a specially designed side and bottom shielding. The availability factor of the SMART plant is 95%, and the occurrence of unplanned automatic scram events is less than one per year.

The major economy improving features of SMART could be summarized as follows: system simplification, component modularization, factory fabrication and direct installation of components at the site, and the reduced construction time. There are some other features contributing to the economy improvement of SMART. For example, SMART uses an advanced on-line digital monitoring and protection systems that increase the system's availability and operational flexibility. The adoption of an advanced man-machine interface technology leads to the reduction of human errors and secures a compact and effective design of the control room with respect to the minimization of staff requirements. The design goals of SMART are shown in Table 1.

TABLE 1. DESIGN GOALS OF SMART

Safety	Core damage frequency	$< 10^{-7}$/reactor-year (RY)
	Radiation release frequency	$< 10^{-8}$/RY
Economics	Electricity generation cost	< Gas turbine
	Construction period	< 36 months
Performance	Availability	> 95%
	Reactor life	60 years

3. COUPLING OF A DESALINATION SYSTEM WITH SMART

The integrated SMART nuclear desalination plant consists of the 4 units that realize multi-effect distillation - thermal vapour compression (MED-TVC) process [4,5], Fig. 2, 3. Each unit has a desalination capacity of 10,000 m^3/day and is coupled with a SMART reactor through the steam transformer. A steam transformer is installed to protect the desalination plant from radioactive contamination.

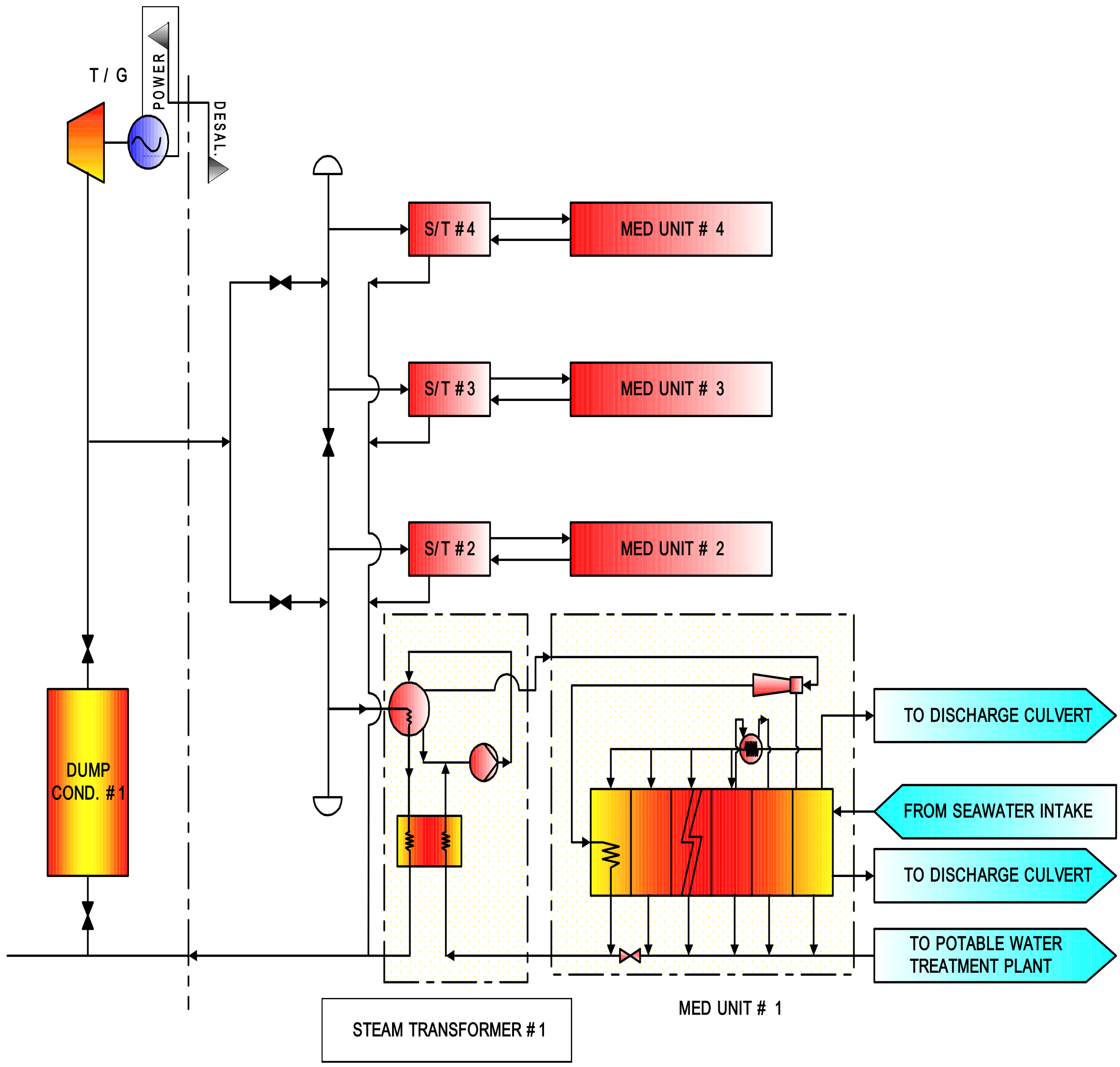

FIG. 2. Schematic diagram of the desalination system.

The steam transformers produce a medium pressure motive steam using the steam extracted from the turbine. The produced steam is supplied through the thermal vapour compressor (steam ejector) to the evaporator for desalination. One significant advantage of the MED-TVC process is its ability to use pressure energy of the steam for desalination. The performance summary of the SMART nuclear desalination plant is as follows:

- Reactor thermal power: 330 MW;
- Design life: 30 years;
- Desalination process: MED-TVC;
- Electricity production: about 90 Mwe;
- Potable water production: 40,000 m^3/day.

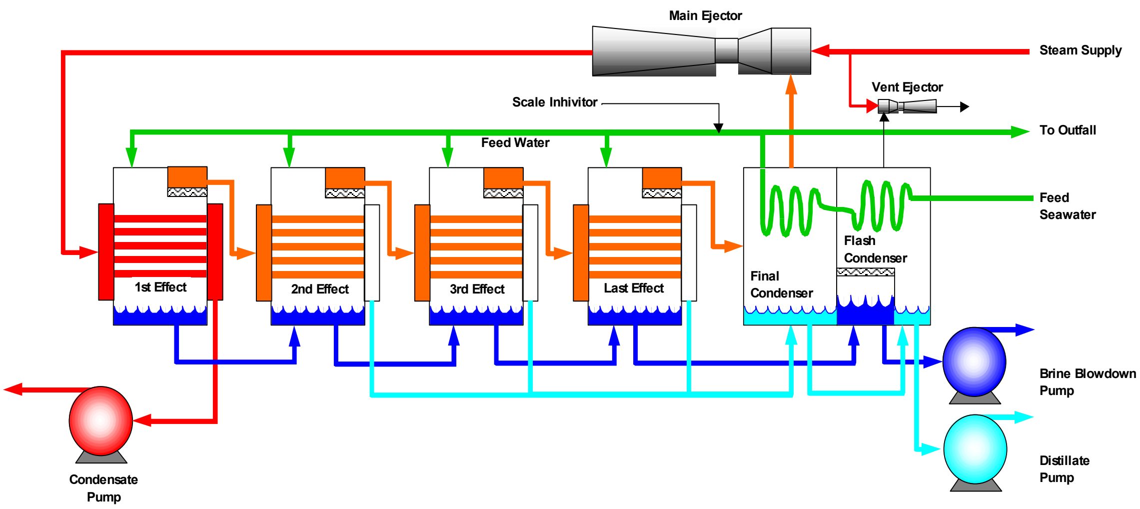

FIG. 3. Schematic diagram of MED-TVC process.

SMART desalination plant incorporates the falling film multi-effect evaporation with horizontal tubes and a steam jet ejector (thermal vapour compressor). The desalination unit is designed with a plant life of 30 years, performance ratio of 19.6, acid cleaning to be performed once in 12 months, maximum brine temperature of 65°C, and the supplied seawater temperature of 33°C. Thermal vapour compressor is introduced to improve thermal efficiency of the process steam. The advantages of this design are high heat transfer coefficients and a relatively simple operation system. The performance ratio of desalination plant, one of the most important coupling parameters, was optimized based on the sensitivity analysis of water production cost and on the requirements to the SMART desalination plant.

The anticipated transients imposed by system interactions between SMART reactor and the desalination plant were identified, and the impacts on safety were evaluated through the bounding approach to key safety parameters. The results of this evaluation show that the key safety parameters do not violate the safety assessment fuel design limits of SMART under any potential disturbance. Therefore, any significant impact on the reactor safety is not expected to come from transients induced by the desalination plant.

In addition, an economic analysis of the desalination plant was performed to investigate economic viability of the SMART desalination plant. The results show that SMART is competitive with other power options, particularly with a gas fired combined plant, within a limited range of electricity generation. The calculated unit cost of fresh water production under desalination capacity of 40,000 m^3/day using the MED process were in the range of 0.56~0.88 $/$m^3$ for 80% plant availability, which is close to the results of studies performed in other countries. These results indicate that SMART can be considered as a competitive choice for seawater desalination.

4. DESIGN VERIFICATION OF SMART

In July 2002, the government of the Republic of Korea has established SMART R&D Centre to take a lead in the development and commercialization of the SMART nuclear desalination plant.

As a principal institution of the SMART project, the SMART R&D Centre is developing a pilot plant to prove the SMART nuclear desalination technology. KAERI together with the Korea Electric Power Engineering Inc. (KOPEC) and the Doosan Heavy Industries & Construction Co. Ltd. are cooperating with the SMART R&D Centre in this project. KAERI is designing the SMART nuclear steam supply system, KOPEC is taking part in the design of the balance of plant, and the Doosan Heavy Industries & Construction Co. Ltd. is in charge of the design activities for the major components of SMART pilot plant and MED-TVC desalination system. The design and construction project for the SMART pilot plant, Fig. 4, has been started in July 2002. The design phase of the project is scheduled to be completed by June 2005, and the plant is planned to be constructed by 2008.

FIG. 4. Plant layout of the SMART pilot plant.

Various transients and accidents were analyzed for the SMART basic design with the computer code developed by KAERI, and the results confirm that safety is ensured with sufficient margins in any postulated transient and accident. The integral effect tests to evaluate the computer code used for safety and performance analysis of SMART is underway at a high temperature and high-pressure thermal hydraulic test facility of 1:96 scale. Figure 5 shows the thermal hydraulic test facility for integral effect tests.

5. INTERNATIONAL COOPERATION FOR SMART PROJECT

In 2002-2003 KAERI together with the BATAN (Indonesia) and the IAEA performed a two-year programme of the “Preliminary economic feasibility study of nuclear desalination in Madura Island, Indonesia” under the framework of IAEA Technical Cooperation Project. The primary objective of this programme was to examine the economic viability of construction of a nuclear desalination plant with SMART in the Madura Island to support industrialization of the Madura Region. Milestones for the construction of a nuclear desalination plant with SMART reactor were established under an agreement between the BATAN and KAERI. The utility requirement documents (URDs) for nuclear desalination in Indonesia are under preparation. Two units of the SMART nuclear power plant are considered to supply 200 MW of electricity and 40,000 tons of potable water per day. The timeline targeted for the plant construction is 2018.

FIG. 5. Thermal hydraulic test facility for integral effect tests.

6. INTERNATIONAL COOPERATION FOR SMART PROJECT

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7. CONCLUSIONS

A nuclear desalination system coupled with the SMART reactor has been developed by KAERI since 1996. The economic feasibility study and safety evaluation of SMART for transients imposed by the interactions with desalination system was completed. A one-fifth scale pilot plant is being constructed to verify performance and safety of the SMART reactor and to demonstrate relevance of the technologies applied for coupling of the nuclear and seawater desalination plants.

An international cooperation programme is in progress involving the Republic of Korea, Indonesia and the IAEA for examining feasibility of the construction of SMART nuclear desalination plants in the Madura Island of Indonesia.

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ANNEX 5

PRELIMINARY FUEL COST EVALUATION FOR CAREM OPERATION WITHOUT ON-SITE REFUELLING

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Abstract. Several small reactors with long life cores (reactors without on-site refuelling) were proposed in the last decade. These proposals were targeted to mitigate concerns related to proliferation resistance and to increase load factor. This paper presents preliminary analysis of fuel costs for the alternative CAREM-300 core configurations that enable the operation without on-site refuelling. The results show that in all cases the fuel costs are generally much higher than for the reference fuel cycle.

1. INTRODUCTION

In the last decade several small reactors with long life cores (reactors without on-site refuelling) were proposed. To address concerns related to proliferation resistance some designs feature a long life straight burn core that eliminates the need to refuel reactor and to reshuffle fuel during the whole fuel lifetime and thus may secure more difficult access to fuel during reactor operation [1].

There is a demand for small nuclear power plants that may suit the needs of developing countries [2] and of some utilities in developed countries. CAREM concept is conceived to target these markets with an offer of 1,000 US$/kWe overnight cost for a 300 MWe nuclear power plant. In this paper, several CAREM-300 core configurations are examined in terms of fuel costs to analyse an option of operation without on-site refuelling.

2. CAREM NUCLEAR POWER PLANT

This section describes the main characteristics of CAREM nuclear power plant [3].

The CAREM design is based on an integrated light water reactor with low enrichment uranium fuel. It is an indirect cycle reactor with some distinctive features that essentially simplify the design and also contribute to high level of safety. The basic design features are [4]:

- Integrated primary cooling system;
- Primary cooling by natural or assisted circulation depending on the module power;
- Self-pressurized primary circuit;
- Passive safety systems.

The primary cooling system is of integrated design: the reactor pressure vessel (RPV) accommodates reactor core, steam generators, primary coolant, and absorber rod drive mechanisms.

For power modules below 150 MWe, the flow rate in the reactor primary system is achieved by natural circulation. Figure 1 shows a diagram of the natural circulation of coolant in primary system. Water enters the core from the lower plenum. After been heated the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces produced by differences in density along the circuit are balanced by the friction and form losses, altogether producing the core flow rate that secures sufficient thermal margin to critical phenomena. Steam generators located above the core boost natural circulation of the reactor coolant, which also acts as neutron moderator.

For power modules over 150 MWe, pumps are used to achieve the flow rate needed to operate at full power.

The steam generators are of a "once-through" type and implement helical tubes. The flows of the primary and secondary systems circulate in a counter-current mode. The secondary fluid circulates upward within the tubes. It flows into the tubes as liquid-water and it reaches the exit as an overheated steam.

Self-pressurization of the primary system is secured by the steam dome and results from the liquid-vapour equilibrium. Large volume of the integral pressurizer also facilitates damping of the eventual pressure perturbations. Due to self-pressurisation, the bulk temperature at core outlet corresponds to saturation temperature under the primary pressure. Therefore, heaters and sprinkles typical of a conventional PWR are eliminated.

The core of CAREM-300, a 300 MWe module, has 199 fuel assemblies of hexagonal section and about 2.85 m active length. Each fuel assembly contains 108 fuel rods, 18 guide thimbles, and 1 instrumentation thimble. Fuel assembly components are similar to those of a conventional PWR. The overall thermal power is 900 MWth, the average linear power is 147 W/cm [5].

Gd_2O_3 is used as burnable poison in specific fuel rods, while movable silver–indium-cadmium absorber rods control core reactivity. The control rod drives placed inside the RPV are hydraulically driven. Liquid boron is not used for reactivity control during normal operation. The design of safety systems meets the regulatory requirements of nuclear industry as for redundancy, independence, physical separation, diversification, and failure into a safe state. CAREM safety systems are designed to eliminate the need of active intervention in accidents within a long period of time.

CAREM has two different and independent shutdown systems that are designed to shut down the reactor and to maintain the required sub-criticality in reactor core. These systems are activated by the reactor's protection system. The first system is designed to shut down the reactor by dropping gravity-driven neutron-absorbing elements into the core. The second shutdown system is based on the injection of borated water into the core, and is gravity-driven also.

In NPP blackout, the residual heat is removed from the core in a passive mode, by natural convection through the residual heat removal system. This system transfers decay heat energy to the pressure suppression pool. CAREM has an emergency injection system to prevent core exposure in the case of a loss of coolant accident (LOCA). This system assures the adequate cool down of reactor's core under deactivated electrical power supply. Three safety relief valves additionally secure the RPV integrity. Having a 100% relief capacity each, they protect the RPV against overpressure.

CAREM has the containment isolation of a pressure-suppression type to retain the eventual release of radioactive materials. It is designed to secure the inside pressure to be below the design pressure in any unprotected LOCA.

3. CAREM AND LONG LIFE CORES

The CAREM-300 standard fuel cycle is based on a 3-batch core refuelling with the initial fuel enrichment of 3.5%. The use of a single-batch core refuelling, higher enrichment and lower power density were considered to analyse fuel costs in operation without on-site refuelling. Using a two-group diffusion model, the discharge burn-ups of fuel were calculated for different cases. The results are shown in Table 1.

TABLE 1. DISCHARGE BURN-UP FOR DIFFERENT CORE STRATEGIES

Initial enrichment [%]	Refuelling zones	Power density [kWth/kg]	Discharge burn-up [MWd/t]	Core life time [full power days]
3.5	3 (*)	35.4	35175	330
3.5	1	35.4	23450	662
3.5	1	17.7	24317	1372
3.5	1	8.86	24750	2793
5	1	35.4	37122	1047
5	1	17.7	38249	2159
5	1	8.86	38813	4381
10	1	35.4	78920	2227
10	1	17.7	80523	4544
10	1	8.86	81324	9179

(*) Reference Case

Preliminary analysis was based on an individual account of each fuel cycle stage to secure that discount rate is properly attributed to each associated cost. Such treatment is necessary since different steps in fuel cycle take place at different time.

The data on costs of fuel cycle stages, taken from [6], is presented in Table 2.

TABLE 2. COSTS OF FUEL CYCLE STAGES

Uranium cost	65 U$S/kg U
Conversion	8 U$S/kg U
Enrichment	110 U$S/SWU
Fuel fabrication	250 U$S/kg U

Different discount rates were applied to represent different regional realities. Figure 1 shows fuel costs for 15% discount rate at various power densities and core enrichments. Such high discount rate may be representative of the situation in a developing country.

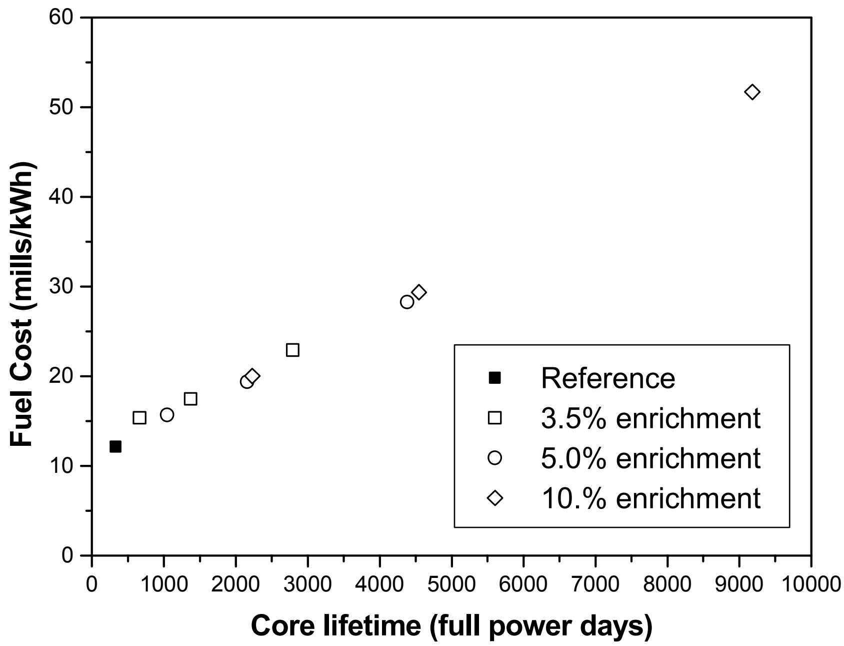

FIG. 1. CAREM-300 fuel costs: 15% discount rate.

To represent the situation in developed countries 10 and 5% discount rates were applied. Figure 2 shows relevant fuel costs for a 10% discount rate. Figure 3 shows fuel costs for a 5% discount rate.

All results indicate that CAREM-300 operation without on-site refuelling would in all cases result in fuel costs noticeably higher than for the reference core with 3-batch refuelling scheme and 3.5% initial enrichment of fuel.

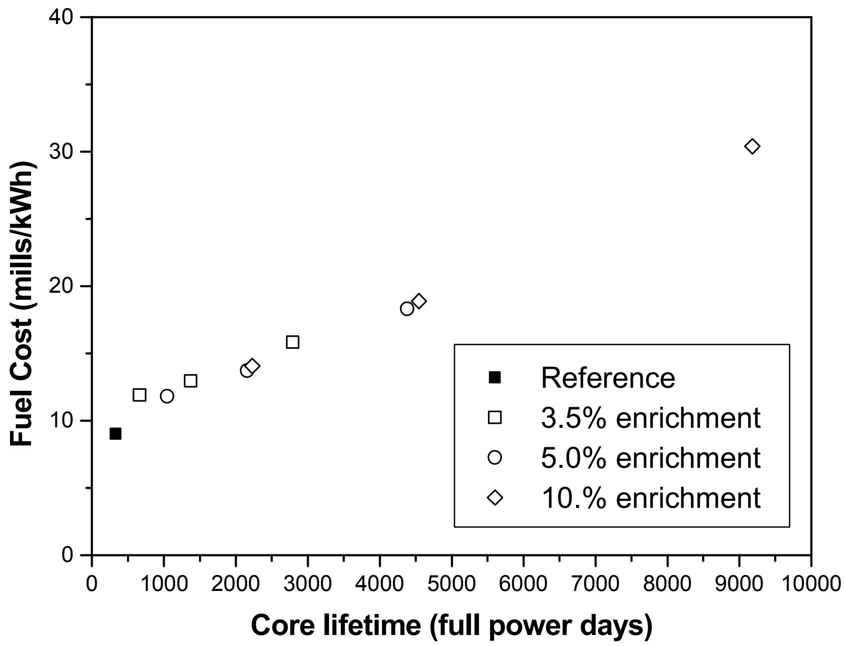

FIG. 2. CAREM-300 fuel costs: 10% discount rate.

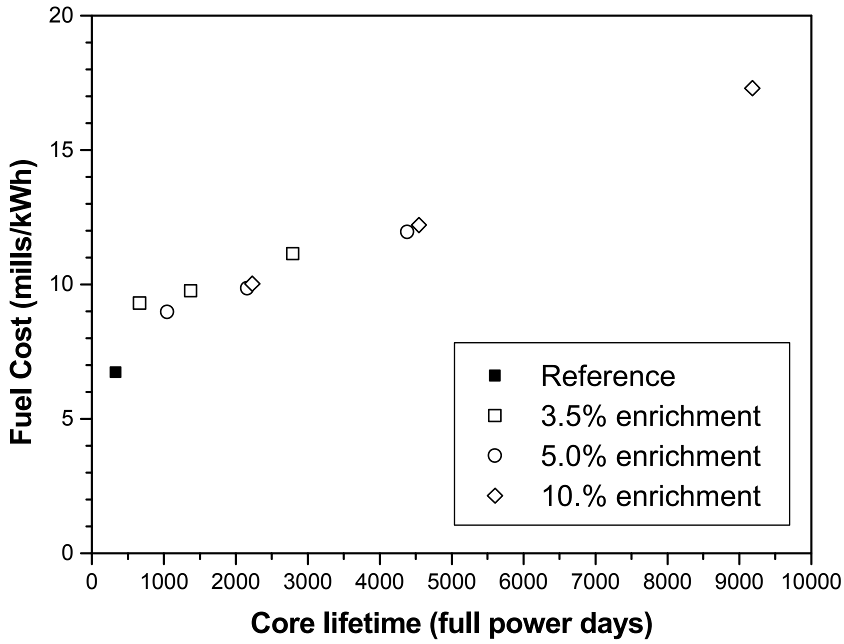

FIG. 3. CAREM-300 fuel costs: 5% discount rate.

4. CONCLUSION

Fuel costs for different CAREM-300 core configurations were analysed. Long life cores were considered but the economical evaluation shows that for CAREM-300 these options are essentially more expensive as compared to the reference core with 3-batch refuelling scheme and 3.5% initial enrichment of fuel.

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ANNEX 6

RUSSIAN CONCEPTS OF NUCLEAR POWER PLANTS WITH SMALL REACTORS WITHOUT ON-SITE REFUELLING

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Abstract. The paper presents an overview of activities for the development of cogeneration nuclear power plants with small sized reactors on-going in the Russian Federation. Small sized nuclear reactors and power plants on their basis represent a prospective class of power sources for the effective solution of several problems such as provision of isolated consumers in many regions of the world with the electric energy, district heating and/or process heat, potable water etc. The economic expediency and social importance of such power units for the provision of heat and power supply to various consumers in the extreme North and Far East regions of Russia, where fuel supplies are costly and unreliable, has been comprehensively justified through a series studies. The paper outlines the criteria developed for the selection of small reactor designs and concepts and provides a technical overview of 15 concepts of NPPs with small reactors currently developed in the Russian Federation.

1. INTRODUCTION

Sparse industrial entities and residential settlements in the far north and far east regions of Russia are traditional customers of small sized power applications. These regions altogether constitute two thirds of the country's territory and are characterized by relatively small power loads concentrated within singular consumption points, by difficulties in the supply and high costs of fossil fuel, and by lack of qualified local personnel needed to maintain complex power equipment. Though only 8% of the population of Russia lives in these regions, more than 60% of the prospected inventory of hydrocarbon fuel and mineral and raw material resources of the country are concentrated there.

As a rule, the use of traditional fossil-fuelled electric power plants and low-capacity boiler plants for heating is associated with the delivery of fuel and combustive-lubricating materials, which is risky and results in the considerable increase of the costs of electric and heat power generated in these regions. Pending problems of power supply became an obstacle of social and economical development of the Russian northern territory [1].

Under such conditions small nuclear power plants are very seriously considered as an ultimate solution of the problem of power supply, which would also enable to solve the associated problems of social and economic development of these regions. The outcome to be achieved through the implementation of small nuclear power plants (SNPP) is defined by the following factors:

- Increased reliability of electric power and heat supply to population;
- Savings on hydrocarbon fuel procurement and supply, and associated reduction of living costs and costs of public facilities;
- Creation of attractive conditions for increase of the efficiency of operating manufactures and for the construction and start-up of new power-industrial centres;
- Reduction of the adverse impacts on local environment, which is characterized by particular vulnerability in these regions.

These factors created a basis to include the provisions for development and deployment of SNPPs into a Special Federal Programme of Russia on "Cost-effective economics … for the period till 2010". The programme provides for construction of "…small nuclear power plants as independent sources of distributed heating and electric power supply in remote regions".

In Russia the possibility to develop SNPPs for the extreme conditions of the far north and east is primarily associated with the use of floating power units (FPU) with advanced marine-type pressurized water reactors [2,3]. Here, the important factors are availability of marine-type reactor designs validated through long-term operation and of proven production technologies for civil ships with such reactors that demonstrated well during operation in the conditions of Arctic. Since 1959, eight nuclear icebreakers have been constructed in Russia and 6 of them are in operation now. Each propulsion unit of an icebreaker has the equivalent electric power of 50 to 70 MW. Also, an ice-grade cargo boat with an NPP was constructed. 13 nuclear reactors of KLT-40 type operated within the propulsion units of these ships, reaching the total accident-free operating time of 250 reactor-years. This experience is being thoroughly used in the development of SNPPs.

The first step to realize the federal programme on SNPPs in Russia is to construct a pilot FPU with two KLT-40 type reactors (KLT-40S). The detailed design of such FPU has been completed; licenses for its construction and placement in Severodvinsk (Arkhangelsk region, Russia) have been obtained [6].

2. CRITERIA FOR SMALL NUCLEAR POWER PLANTS (SNPPs)

The following criteria were used for assessment/selection of the designs (concepts) of small reactors for SNPPs:

- Ability to operate without refuelling and reshuffling of fuel during a sufficiently long period selected taking into account the factors of plant economy and safety. Such mode of operation eliminates the necessity to store fresh and spent fuel outside the reactor. Refuelling and storage could be accepted if the used mode of nuclear materials protection excludes their unauthorized use;
- Difficult unauthorized access to fuel during the whole period of its presence at NPP site and transportation;
- Design measures to facilitate implementation of IAEA safeguards;
- Very high level of safety consistent with the anticipated scope of global expansion of such reactors;
- Ability to cope with all postulated beyond design accidents caused by natural and human-induced internal events without unacceptable radiation consequences for the population and with preserving the possibility to return the reactor to a manufacturer country.

Additional features and characteristics were assumed desirable for reactors of the considered type:

- Factory fabrication and fuelling that secures the supply of a reactor module being reliably protected against unauthorized access;
- Fuel load that is unattractive for purposes of a weapon programme;
- Improved competitiveness and high fabrication quality due to design standardization, mass production and common basis for licensing;
- Diversity of applications, including electric power generation, co-generation of electric power and heat, production of potable water and hydrogen, etc;
- A variety of siting options, including those near the populated centres, as well as in remote and hard-to-access areas, scattered islands, etc;
- Simplified reactor control, tolerance to human errors;
- Minimum reliance on sophisticated local infrastructure.

Based on the above criteria, the following designs/concepts of small reactors could be recommended for SNPPs: SAKHA-92, ABV, KLT-40S, RIT, VBER-150, all developed by OKBM (Nizhny Novgorod, Russia). All of them are based on a well-established reactor fabrication technology and implement a broad knowledge base thoroughly rooted in the experience of Russian marine reactors that have an operation experience of more than 6000 reactor-years [5,7]. The innovations in these designs are related, first of all, to the realization of a floating version of power unit, i.e., the application of radically changed system configuration. The demonstration prototypes of such floating power units (FPUs) can be deployed in Russia within the next 5-8 years, this term being defined mainly by availability of adequate financing. They also have a considerable potential for further improvement in line with the defined criteria. Such improvements could be realized gradually within the next decade, and would result from step-by-step advancements in both nuclear reactor and nuclear fuel cycle technology.

In their functional designation, all mentioned above designs (concepts) are targeted at the generation of electricity and/or production of heat. The designs of systems for co-generation and seawater desalination have been developed, for example, for potable water production a combination of evaporation-type technology (MSF) and reverse-osmosis technology (RO) has been mastered [8,9]. Russian and foreign manufacturers of desalination equipment participated in this activity [10].

One of the key requirements to reactors of SNPPs is the ability to operate without refuelling during a sufficiently long period, which results in the elimination of a necessity to store fresh and spent fuel at NPP site outside the reactor. Some options to meet this requirement are available, for example:

(a) The approach realized in SAKHA-92 reactor plant assumes that reactor core lifetime is equal to the reactor plant lifetime. After NPP decommissioning, spent fuel is transported in special protection containers to the reprocessing and disposal site. One of the demerits of this option is that it could be realized for very small plants only: the power of SAKHA-92 plant is 1 MW.

(b) Another option is to use cores with the lifetime equal to the period of plant operation between factory repairs and maintenance. This option can be realized for FPUs, which are periodically transported for repair to a specialized factory, at which the spent fuel is discharged from the reactor also. Such approach is realized in ABV type reactor plants. The maximum power of nuclear cogeneration plant based on two ABV-6M reactors is limited by the value of 12 MW, which fits into the range of 10 to 25 MW considered as having the highest market potential.

(c) Next option is based on the use of special maintenance ships that provide for fresh fuel delivery immediately prior to the start-up of refuelling and for spent fuel removal immediately after the refuelling completion. Such ship accommodates refuelling equipment and trained personnel and can provide services to several FPUs. This scheme is already realized for Russian ships with nuclear propulsion.

(d) Finally, there is an option to use an FPU with incorporated spent fuel storage. Spent fuel remains in this storage till the completion of the period between factory repairs. After that, FPU is transported to the point of repair and spent fuel discharge. This solution is adopted in nuclear cogeneration plant with KLT-40S reactor for Severodvinsk town.

Refuelling, storage and transportation operations in the last two options should be performed in such a way as to eliminate unauthorized use of spent fuel. Realization of these options makes it possible to increase maximum power of the plant significantly. Both of these options are available for KLT-40S, VBER, and RIT designs upon a request from the customer. In this, increase of core lifetime is one of the key directions for improvement of each of these plants.

More details on the features and characteristics of the abovementioned designs (concepts) are provided below.

3. SAKHA-92 NUCLEAR COGENERATION PLANT

The small nuclear cogeneration plant SAKHA-92 [11] is a small-size power source intended for generation of electric power and district heating. The maximum electric power supplied to the consumer is 1000 kW. Low-grade heat output falls in the range of 1200 to 3000 kW at electric load drop.

SAKHA-92 is a maintenance-free nuclear power plant of increased safety. Plant design was developed on the basis of PWR technology, but implements integrated steam and gas pressurizer systems and relies on natural circulation of the primary coolant (Fig. 1). The use of such designs as leak-tight turbine-generator, canned condensate and feed pumps allows to secure the tightness of both primary and secondary circuits, which in turn make it possible to exclude some auxiliary systems.

Safety concept of SAKHA-92 nuclear cogeneration plant is based on inherent safety, as well as on successive functional and physical barriers between nuclear fuel and environment. Integral design, natural circulation of primary circuit coolant, and negative reactivity coefficients secure the operator-free scheduled and emergency modes of operation within a certain range of plant parameters.

The reactor is arranged in a strong-tight guard vessel. The use of passive safety systems excludes radioactivity release beyond the vessel boundary and secures bringing the reactor to a safe state in all possible modes of operation.

The power unit is transported to a site in separate modules, all factory-set. Different modes of transportation can be used: by railway, by sea, or by truck.

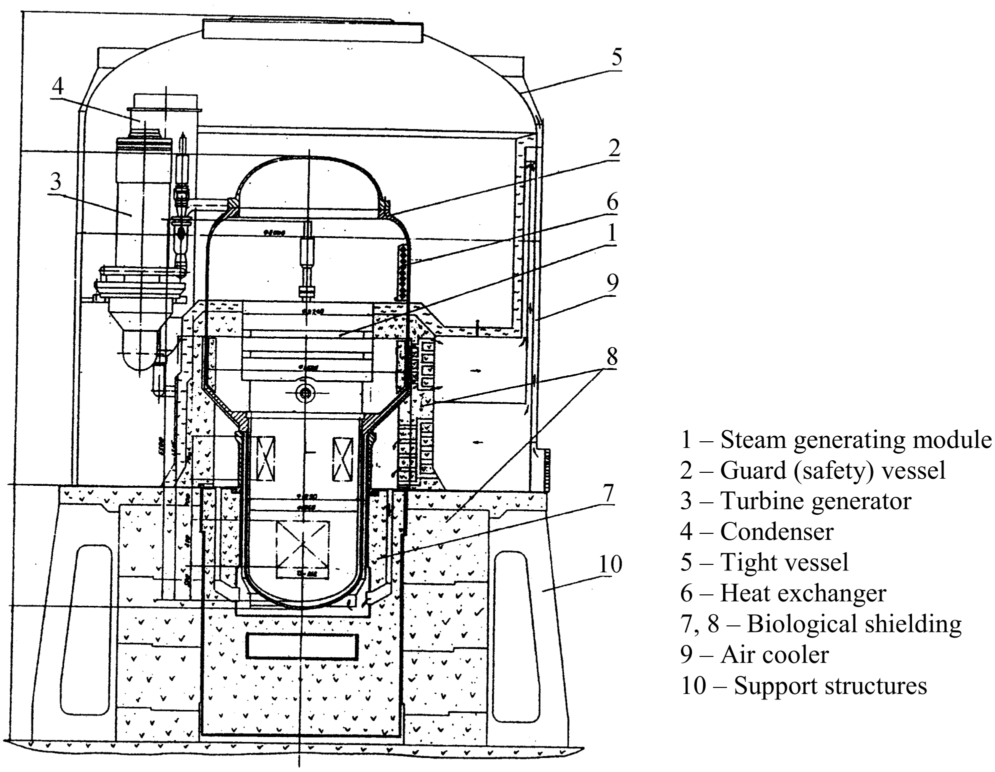


FIG. 1. NPP with SAKHA-92 reactor.

The plant service life is 25 years, which corresponds to the period of operation without on-site refuelling and to the period of major equipment replacement. After the end of service life, spent fuel is discharged from the reactor and the plant is disassembled and transported to a disposal site. Floating nuclear cogeneration plant based on SAKHA-92 was considered also.

4. ABV POWER PLANT AND FLOATING POWER UNIT BASED ON ABV REACTORS

The ABV reactor plant [2,7,16,17] is an integral pressurized water reactor with natural circulation of coolant in all-operational modes (Fig. 2). It is a reactor highly compact and stable to any internal and external impacts. High level of safety is achieved through strong reliance on inherent safety features. The ABV reactor has simplified equipment and systems and a compact protective shell (Fig. 3). Rated thermal power depends on the configuration and falls within the range of 16 MW (ABV-3) to 38 MW (ABV-6). Floating nuclear cogeneration or district heating plants with ABV reactors could be created, offering a wide range of variations in the electric power (from 2.5 to 12 MW) and thermal output (from 20 to 100 GJ/h) as shown in Fig. 4. The reactor design incorporates passive and self-actuated safety systems. The displacement of an FPU with ABV reactors is shown to be relatively small, therefore the units may be supplied all assembled.

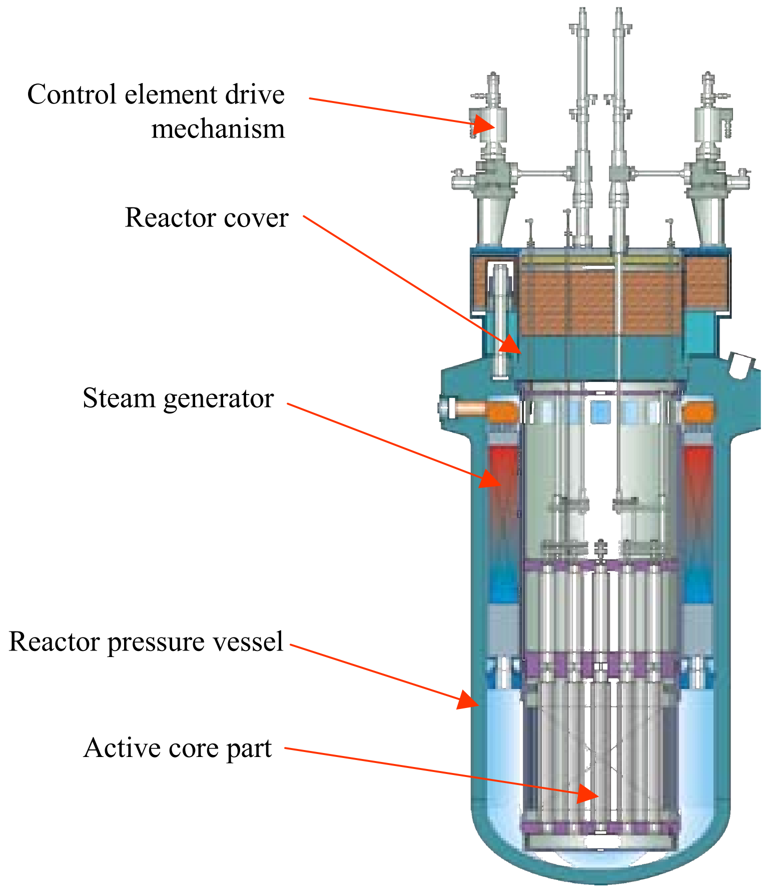


FIG. 2. ABV-6 reactor.

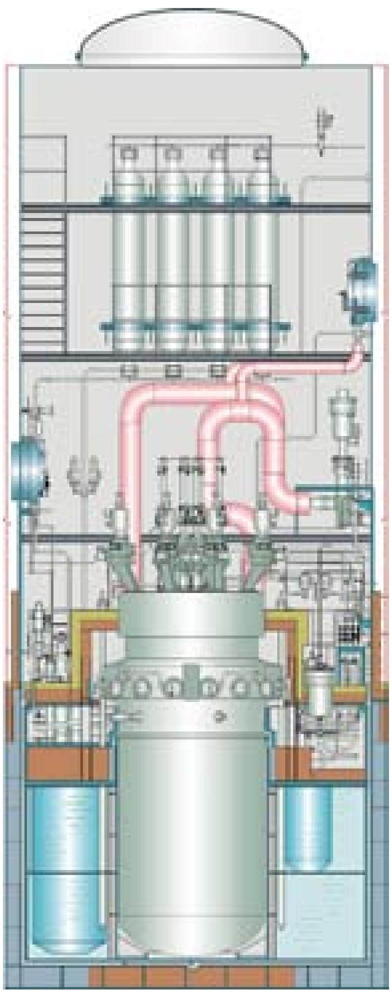


FIG. 3. NPP with ABV-6 reactor: inside containment structures.

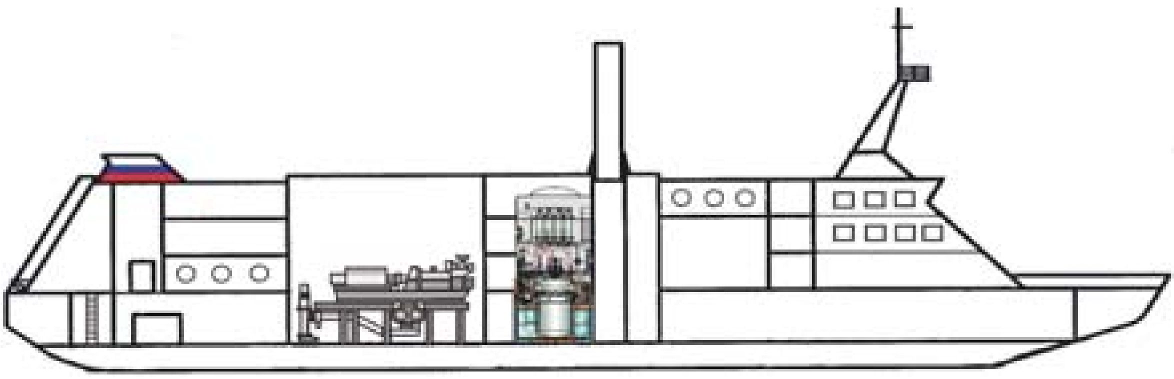


FIG. 4. Floating nuclear cogeneration plant with ABV reactor.

The OKBM (Nizhny Novgorod, Russia) developed a detailed design of the ABV-6 basic reactor plant, has performed R&D for the validation of this design, and continues to look for improved design solutions. Recent design studies prove the possibility to increase thermal power of the core up to 45 MW and core lifetime up to 10-12 years, consistent with the period between FPU repairs. Uranium enrichment is less than 20%, which meets the IAEA recommendations on non-proliferation. Major characteristics of FPUs with ABV reactors are presented in Table 1.

TABLE 1. CHARACTERISTICS OF FLOATING NUCLEAR COGENERATION PLANTS WITH ABV REACTORS

Characteristic	Floating nuclear cogeneration plant with ABV-6		Floating nuclear cogeneration plant and nuclear district heating plant with ABV-3	
	Double-unit plant	Single-unit plant	Single-unit plant	
Electric power, MW	12	6	2.5	- *
Heat output, GJ/h	100	50	26	53
Length, m	108	80	67	64
Width, m	14	14	12	12
Draught, m	2.6	2.3	2.0	2.0
Displacement, t	3,700	2,300	1,600	1,500
* Heat supply only				

The reactor refuelling is performed at the factory, being combined with the FPU factory repair. The economic characteristics of FPUs with ABV reactors are given in Table 2.

TABLE 2. CHARACTERISTICS OF FLOATING NUCLEAR COGENERATION AND DISTRICT HEATING PLANTS WITH ABV REACTORS

Characteristic	Floating nuclear cogeneration plant with ABV		Floating nuclear cogeneration plant and nuclear district heating plant with ABV-3	
	Double-unit plant	Single-unit plant	Single-unit plant	
Plant construction cost, million $	63.9	36.4	27.2	22.4 (nuclear district heating plant)
Construction period, years	5	4	4	4
Prime cost of electricity, cent/ kWh Prime cost of heat, $/GJ	8.0 4.5	8.0 4.8	12.0 7.0	- 7.5
Payback period, years	4-5			

5. KLT-40S POWER PLANT AND FLOATING POWER UNITS BASED ON KLT-40S REACTORS

At present, more progress is observed in the activities for a floating nuclear cogeneration plant based on KLT-40S reactors [13] - see Fig. 5. So far, the basic design of this FPU has been developed and approved, licenses for its construction and accommodation in Severodvinsk (Arkhangelsk region, Russia) have been obtained.

Floating power unit has been developed to produce electricity and heat and to transfer them to customers making use of the coastal infrastructure. Safe positioning and retaining of FPU is provided by the hydraulic-engineering structures. The coastal infrastructure includes structures and special devices for the reception and transmission of electric power and heat to users, and is operated co-jointly with an FPU.

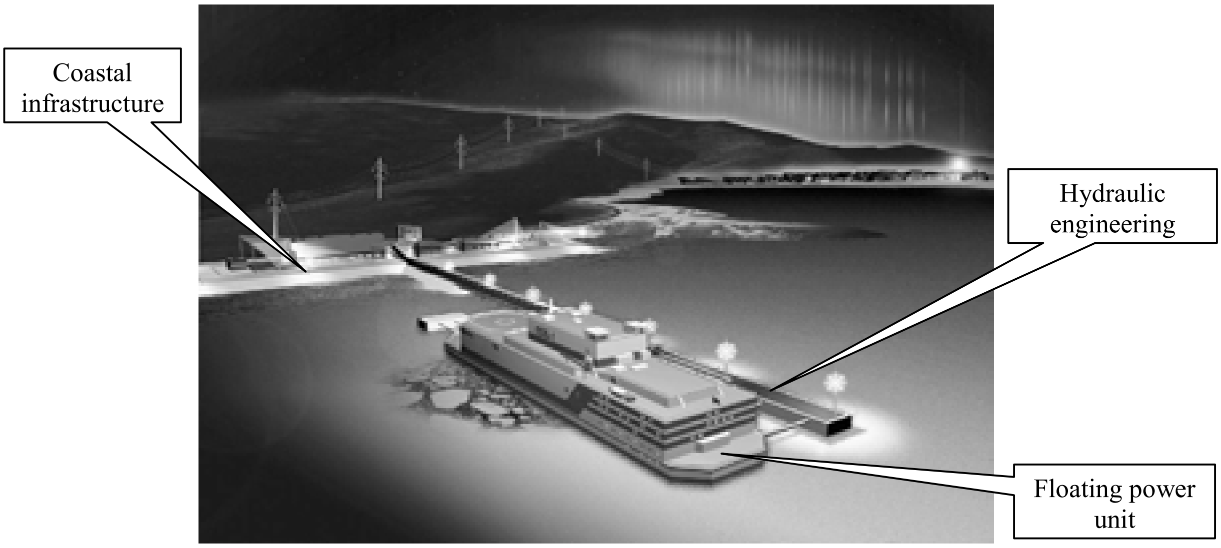


FIG. 5. Overall view of a floating nuclear power unit with KLT-40S reactor.

Floating power unit with KLT-40S

The FPU is a smooth deck non-self-propelled ship with the following characteristics: 140 m length, 30 m width, 5.6 m draught, and 21,000 t displacement. The FPU consists of a living module and a power module (Fig.6). The power module accommodates two KLT-40S reactors, two steam turbine plants and electric power system. The nominal electric power supplied to outside consumers is 60 MW and the corresponding thermal power is 146 GJ/h [14].

The FPU houses spent fuel assemblies, storage for solid and liquid radioactive wastes and a facility for reactor refuelling. All operations with radioactive materials are performed only within an FPU, and its design provides five isolation barriers to exclude inadmissible radioactive releases to the environment (Fig.6).

The FPU is manufactured at a specialized shipyard factory and transported to an operation site fully assembled. For repair, the FPU is transported to a specialized repair plant. After the end of its service life, the plant is transported to disposal site (factory), securing that no radiation and other consequences are present in the region of its operation.

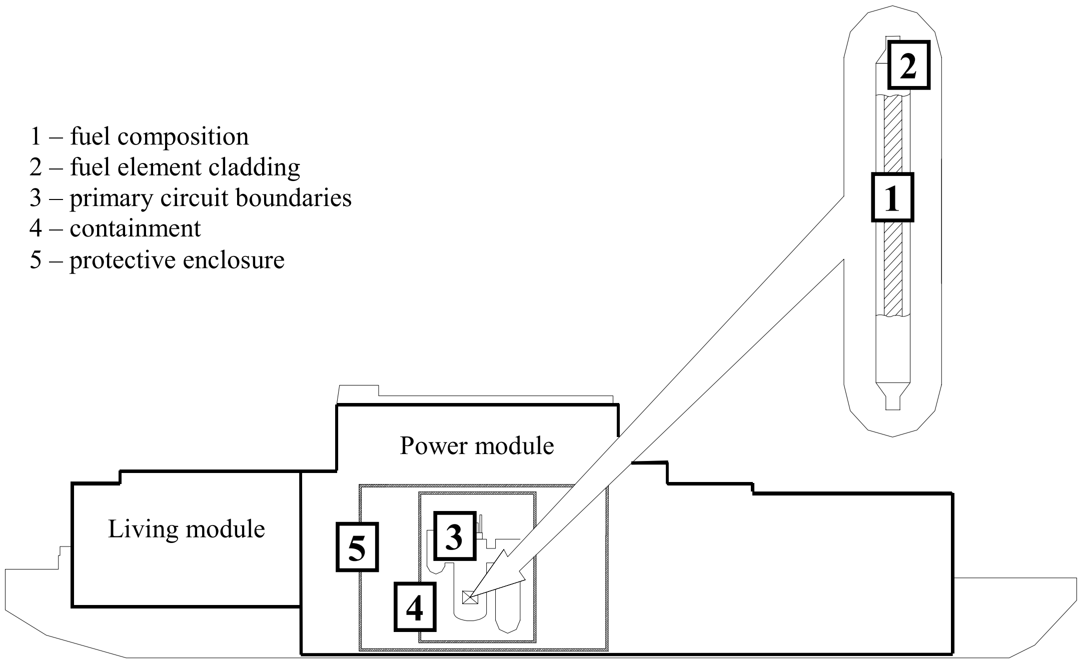


FIG. 6. Localizing barriers of a floating power unit.

KLT-40S power plant

The KLT-40S power plant [13] was developed on the basis of a standard KLT-40 type nuclear propulsion plant that has the experience of more than 250 reactor-years of failure-free operation. Components of the original plant have been modernized to increase plant reliability, to extend its service life and to improve the conditions of maintenance. The design of safety systems is based on safety regulations for marine reactors and was updated to meet the requirements of the Russian Regulatory Authority – GAN RF - for nuclear power plants.

The major characteristics of KLT-40S power plant are as follows:

- Reactor thermal capacity	150 MW
- Steam generating capacity	240 t/h
- Service life	35- 40 years
- Inter-repair period	10-12 years

6. VBER-150 POWER PLANT AND FLOATING POWER UNIT BASED ON VBER-150 REACTORS

VBER reactor plant is developed by OKBM with reliance on the experience of development, construction and operation of the modular nuclear propulsion reactors that are well established and proven through long-term successful operation within civil and navy ships. At present, the total accident-free operating time for them exceeds 6000 reactor-years.

Modular arrangement of the main reactor components represents a key factor of the reactor design (Fig. 7). The reactor pressure vessel, two once-through steam generators and two main circulating pumps are integrated into a single-vessel system. The reactor plant represents a two-loop modification of the more powerful VBER-300 reactor [18,19] that is being developed by OKBM for medium sized nuclear cogeneration plants. The capacity of VBER-150 is about 110 MWe.

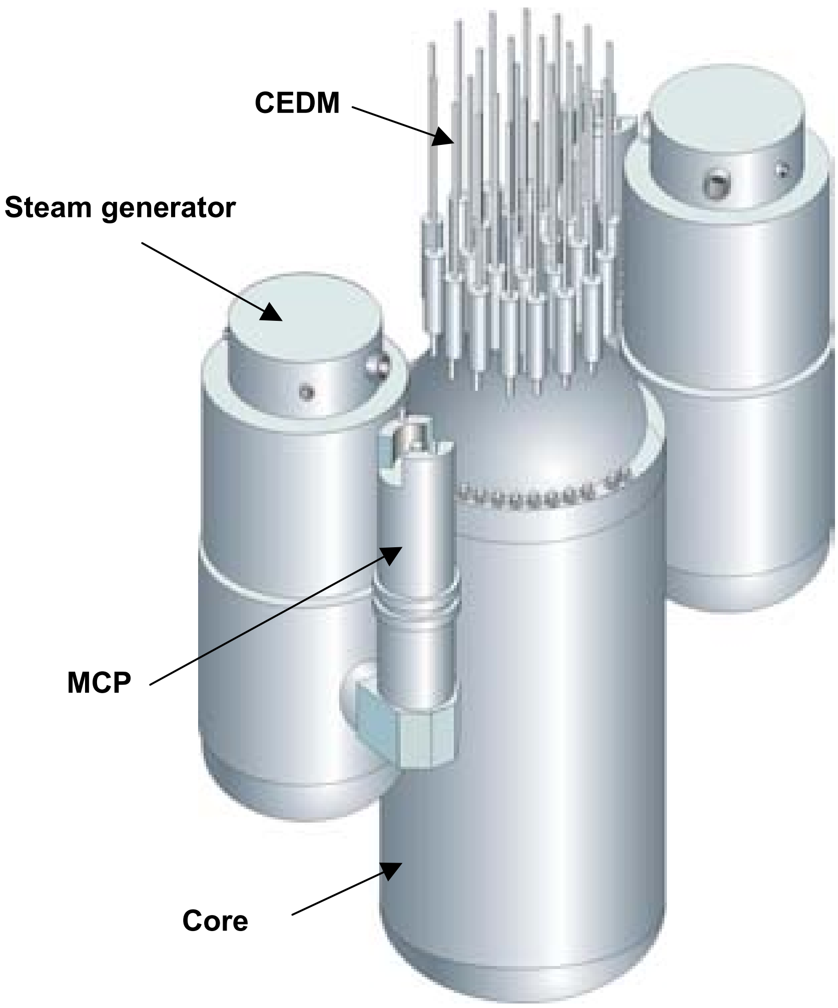


FIG. 7. VBER-150 reactor module (CEDM is control element drive mechanism, MCP is main circulation pump).

VBER-150 combines basic engineering solutions originally developed for nuclear propulsion reactors and for recent designs of NPPs with VVER-1000 reactors, and is capable of meeting all essential safety, reliability and efficiency requirements currently considered for new generation power plants.

The core with increased lifetime is a principal feature of the design. Ductless AFA-type fuel assemblies with rigid skeleton structure developed by OKBM are used in the core. AFA-type fuel assemblies have high load bearing capacity and are resistant to deformation. The engineering solutions for VBER fuel assembly and core structure are proven by positive operating experience of VVER-1000 reactors. They provide a core lifetime of about 50,000 effective hours and the refuelling interval of 7 to 8 years.

Improved fuel design with reduced thickness of fuel cladding and enlarged diameter of fuel pellet currently developed for VVER reactors will also enable the VBER core lifetime to be extended. This would make it possible to ensure VBER-150 operation without on-site refuelling during the whole period between FPU repairs.

Another important feature of FPU with VBER-150 is that reactor plant and steam-turbine plant are arranged on separate floating entities.

Main technical characteristics of FPU with VBER-150 reactor are presented in Table 3.

TABLE 3. CHARACTERISTICS OF FPU WITH VBER-150

Characteristic	Value
Length, m	105
Width, m	17
Board depth, m	7.6
Draught, m	3.6
Displacement, t	~5900
Capacity, MWe	100-110

7. CONCEPTUAL DESIGN OF INTEGRAL REACTOR (RIT) FOR FLOATING POWER AND DESALINATION COMPLEX

In recent years OKBM in cooperation with Russian enterprises under the leadership of the State Scientific Centre of the Russian Federation IPPE (Obninsk, the Russian Federation) has carried out multi-variant conceptual design studies of floating nuclear power and desalination complex (FNPDC). Options considered for FPNDC were different in the number and types of reactor and turbine plants, desalination plants and their interface circuits. The Design Bureau “Lazurit” (Russia) has developed one of the most prospective options for FPNDC. It is based on an integral reactor design RIT.

The reactor plant with RIT reactor of 150 MW thermal is being developed by OKBM on the basis of the design experience of such integral reactors as ABV, AST-500, ATETS-80, ATETS-200, VPBER-600 [15-17, 20-23].

The design of RIT is shown in Fig. 8. The overall dimensions of the reactor unit and the protective shell were essentially reduced. The reactor design was essentially simplified. The operation period was increased by 3.5 times as compared to KLT-40S reactor. The adopted engineering solution resulted in a significant reduction of the probability of LOCA and in simplification of the emergency core cooling systems.

The reactor unit is factory assembled and transported to a shipyard by railway.

New design of the “barge” based on a combination of floating platforms (Fig. 9) made it possible to develop FPU of 5500 t displacement and 2.8 m draught. Such FPUs could be built in shipyards and transported by river. The shipyards are also capable of building floating desalination units (FDUs) of 100 000 t/day output [12,23,24].

FPU and FDU are tugged to the operation area all set. Then they are moored to the equipped berth and connected with each other and to the coastal communications. In the coastal area there are buildings and structures that receive electric and heat power from FPU and desalinated water from FDU and distribute them to coastal grids and networks. On-site storage of any radioactive materials on is excluded.

Being transportable, FPU is periodically towed to a special overhaul base for repair and refuelling.

The performed technical and economic analyses have shown that floating nuclear power and desalination plants with KLT-40S and RIT reactors are competitive with the plants based on fossil fuel, if the fossil fuel price is above 55 $ per ton of standard fuel [12,24].

The following goals were defined for further improvement of FNPDC economic performance:

- Reduction of the number of refuellings;
- Increase of the period between repairs;
- FPU service lifetime increase up to 50 - 60 years;
- Increase of availability factor through optimization of FPU design and operation.

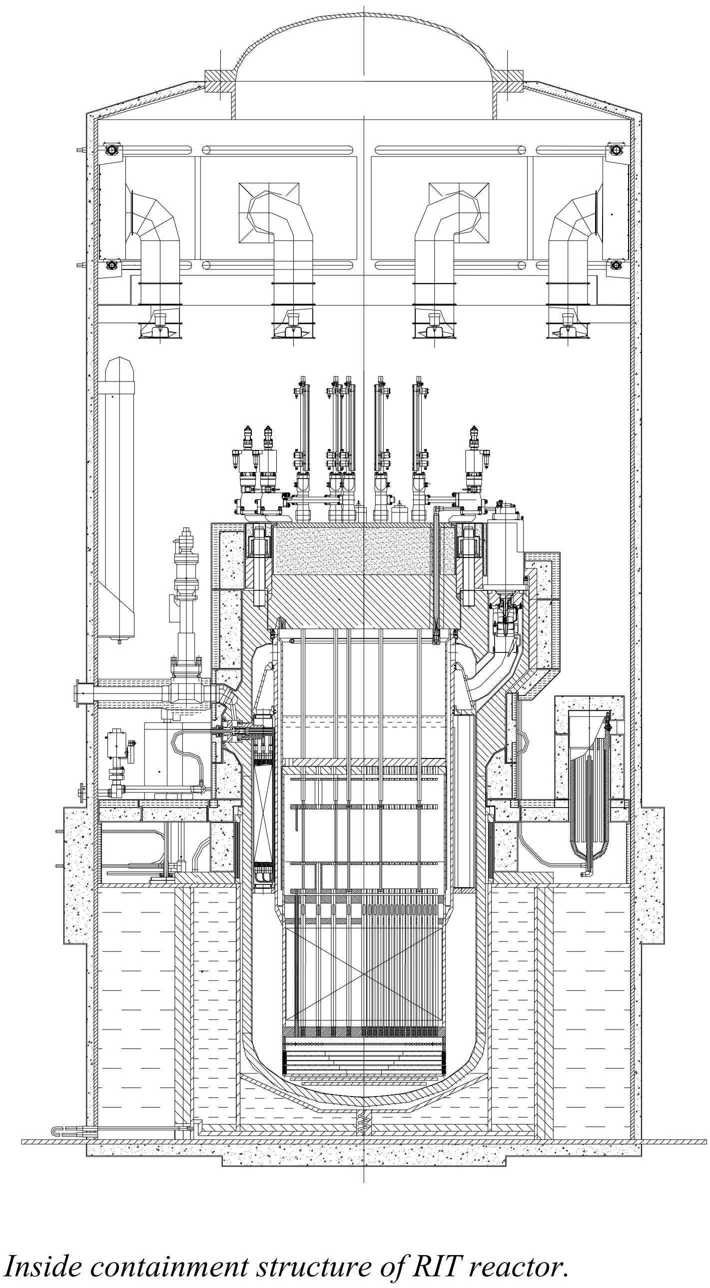
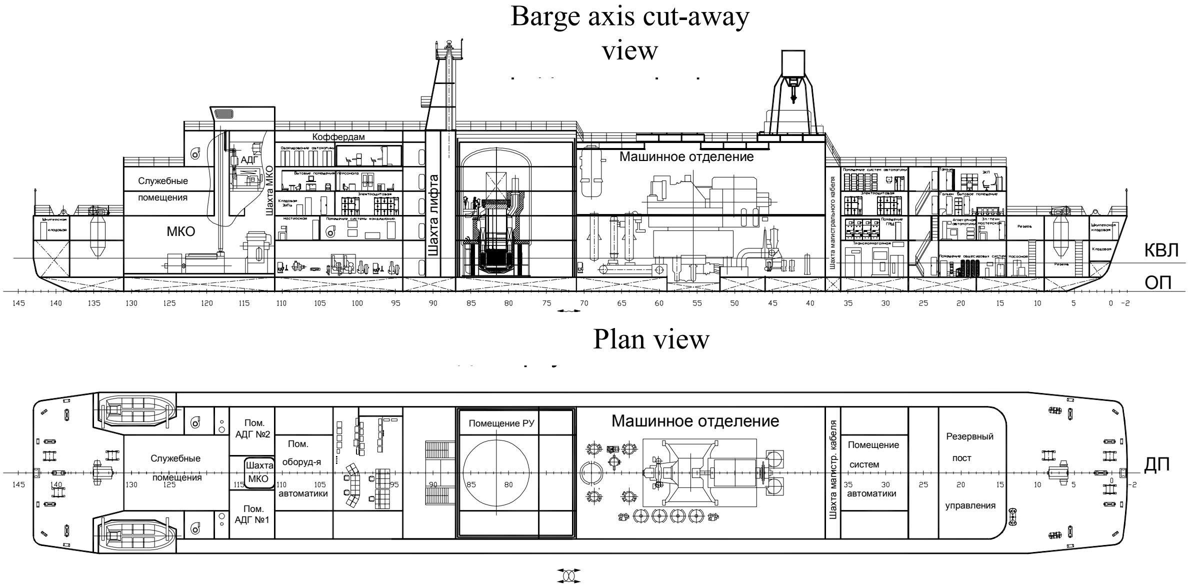


FIG. 8. Inside containment structure of RIT reactor.

FNPDC with single-reactor FPU and desalinating unit of 100 000 m^3/day output was shown to provide better economy, while being suitable for many potential markets worldwide.

In parallel with the ongoing design activities by OKBM, other Russian designers are developing several designs of small nuclear power plants. The outline of these activities is given in the following chapters.



Key data:

Overall length, m – 115
Middle width, m – 17
Vessel side height, m - 8
Draught, m – 2.8
Displacement, t – 5500

FIG. 9. Floating power unit with RIT reactor.

8. CONCEPT OF MARS REACTOR (REACTOR WITH MICRO FUEL ELEMENTS AND MOLTEN SALT COOLANT) AND POWER PLANTS ON ITS BASIS

The Russian Research Centre “Kurchatov Institute” (RRC KI) has developed the conceptual design of an integral reactor of 16 MW thermal with its core consisting of spherical fuel elements similar to those used in high temperature gas cooled reactors (Fig. 10), but being cooled by molten salt coolant [25-29]. Two variants of the core design for 15 and 60 years of operation without on-site refuelling have been developed.

The coolant (a mixture of eutectic compounds) has high boiling temperature (~1300°C) at low pressure and freezes when it gets outside the reactor vessel. For electric power generation, an effective air-turbine cycle is used, making no use of water as heat receiver.

Small NPPs with MARS reactor are developed as autonomous sources for electric power co-generation (up to 6 MW) with high-grade and low-grade heat production (up to 8.5 MW) and seawater desalination. Different options for nuclear cogeneration plant are considered: floating, ground based, or underground.

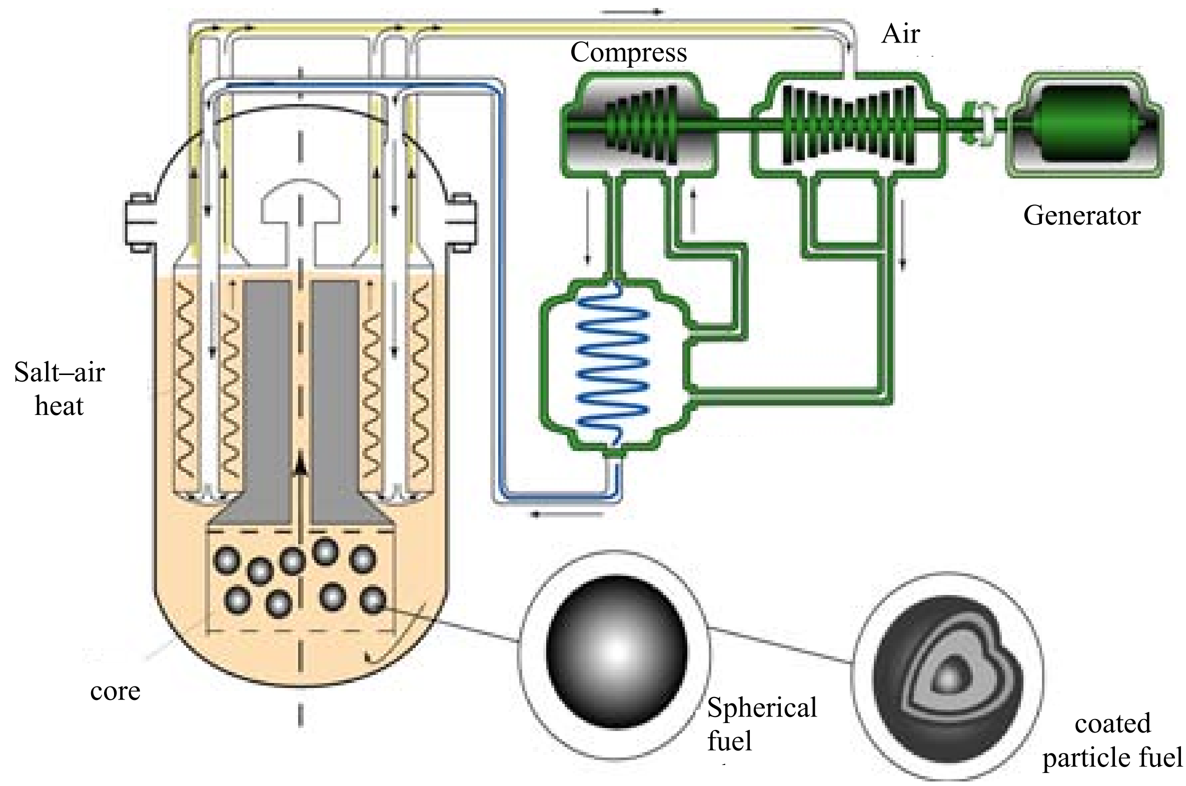


FIG. 10. Principal scheme of MARS reactor plant.

9. CONCEPT OF PWR WITH PEBBLE BEDDED MICRO FUEL DIRECTLY COOLED BY BOILING WATER

The Federal State Enterprises NIKIET and VNIIAM have developed conceptual design of a boiling water reactor with pebble bedded micro fuel and of a 300 MWe nuclear cogeneration plant VKR-MT [30-32], based on a concept proposed by RRC KI. The features of this design are defined both by properties of pebble bedded micro fuel elements and the characteristics of boiling water coolant.

The selected diameter of micro fuel element, which is 1.8 mm, secures an exceptionally high heat exchange surface in the core, which results in a considerably lower heat flows and temperatures as compared to conventional BWRs or PWRs. The low thermal energy stored in the core secures high level of intrinsic safety in design basis and beyond design accidents.

The selected variant of pebble bedded micro fuel elements considerably increases the volume fraction of fuel in the core (up to 30%) and owing to it provides an option of reactor operation without on-site refuelling over a period of 10 to 15 years with the initial enrichment of uranium fuel being below 20%.

This concept also preserves all traditional advantages of boiling water reactors as compared to pressurized water reactors, such as reactor self-regulation by pressure, temperature, and steam content, lower coolant pressure in primary circuit, considerably lower coolant leakage rate at pipeline rupture, and others.

10. MULTIPURPOSE SMALL FAST REACTOR SVBR-75/100

The integral layout of primary pool-type circuit equipment with complete exclusion of valves and pipelines is realized in the design of a 75 to 100 MWe modular fast reactor with liquid-metal Pb-Bi coolant, SVBR-75/100, developed by the State Scientific Centre of the Russian Federation IPPE [33,34], as shown in Fig. 11.

The reactor mono-block, all as a set, could be transported by railway, automobile or water transport, with its core being filled with the consolidated coolant. It is also planned to store spent fuel in leak-tight boxes filled with consolidated Pb and cooled by air from outside.

The conceptual design of a two-unit NPP based on 16 SVBR-75/100 reactor installations and a single turbine plant of 1600 MWe has been developed. In particular, this design is proposed for the renovation of NPPs with nuclear steam supply systems that are approaching the end of their design lifetime.

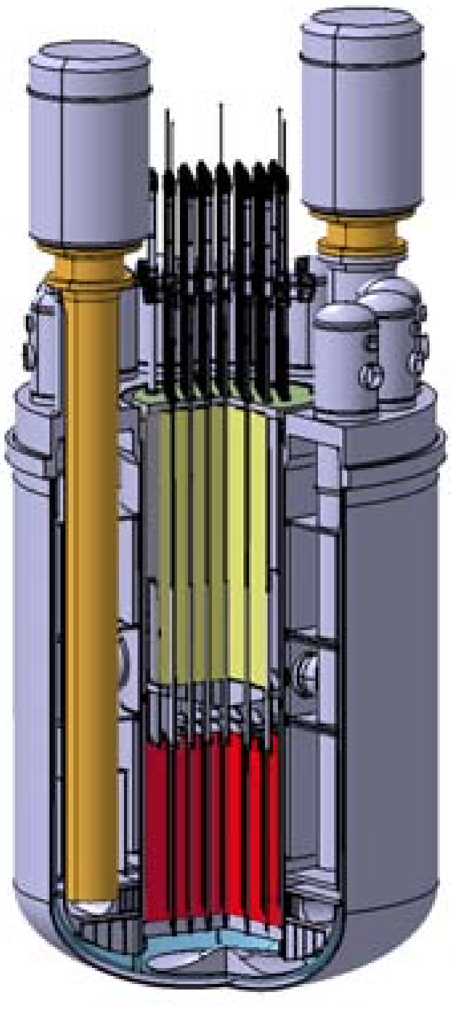


FIG. 11. General view of SVBR-75/100 reactor module.

11. CONCEPT OF UNATTENDED-OPERATION NUCLEAR COGENERATION POWER PLANT ELENA

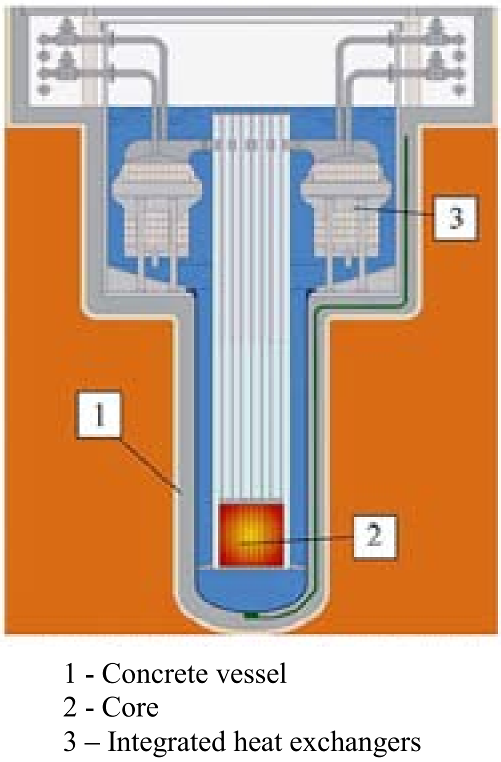
RRC KI proposed a concept of nuclear cogeneration plant ELENA [35] intended for long-term unattended electric power cogeneration (up to 100 KW) with heat production for district heating (up to 3 MW). The design is based on an integral reactor located in the large volume of secondary water. Electric power is generated in semiconductor thermal battery due to the temperature difference provided between primary and secondary circuits. Natural circulation of coolant in both circuits ensures the NPP unattended operation without on-site refuelling for up to 25 years).

The GAMMA testing facility, which is an experimental reactor with thermal-electric conversion of nuclear energy and a prototype of the ELENA plant, has been operating at RRC KI for about 15 years.

12. LOOP-TYPE REACTOR RUTA-70 FOR NUCLEAR DISTRICT HEATING PLANT

The Federal State Enterprise NIKIET has developed conceptual design of an NPP based on the reactor of 70 MW thermal, which appears as a stationary concrete pool covered from inside by stainless steel, in the lower part of which the core is located (Fig. 12). Natural or forced circulation of primary coolant is provided [36].

Low parameters of primary circuit, such as pressure close to the atmospheric and temperature only slightly higher than 100ºC secure high reliability, intrinsic safety and ecological compatibility of nuclear district heating plant based on RUTA reactor intended for district heating of settlements. Altogether, these features make it possible to construct such plants in maximum proximity to the consumers.



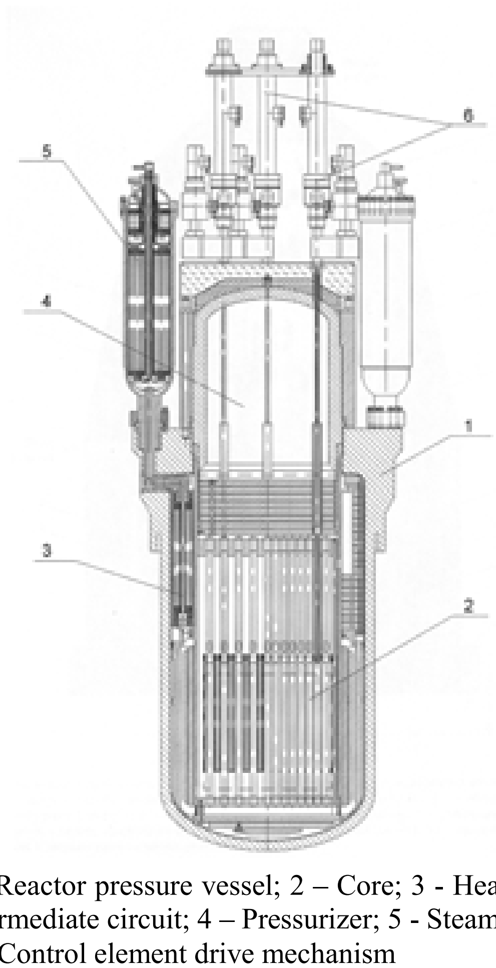
1 - Concrete vessel
2 - Core
3 – Integrated heat exchangers

FIG. 12. RUTA-70 loop-type reactor.

13. TRANSPORTABLE AUTONOMOUS NUCLEAR POWER PLANT WITH UNATTENDED INTEGRAL REACTOR UNITERM

The Federal State Enterprise NIKIET has developed a conceptual design of a cogeneration plant with an integral modular PWR type small reactor UNITHERM [37] - see Figs 13, 14. The design assumes that fabrication, assembly and balance and commissioning of certain NPP modules are performed at specialized machine-building Enterprises, with only a small number (10-15 pieces) of large modules (from 100 to 175 tons) being supplied to the site. The principal characteristics are:

Electric power for consumers	1.5 MW
Thermal power for consumers	4.0 GCal/h
Period of operation without on-site refuelling	20 years



1 - Reactor pressure vessel; 2 – Core; 3 - Heat exchanger of intermediate circuit; 4 – Pressurizer; 5 - Steam generator; 6 – Control element drive mechanism

FIG. 13. UNITERM-70 reactor.

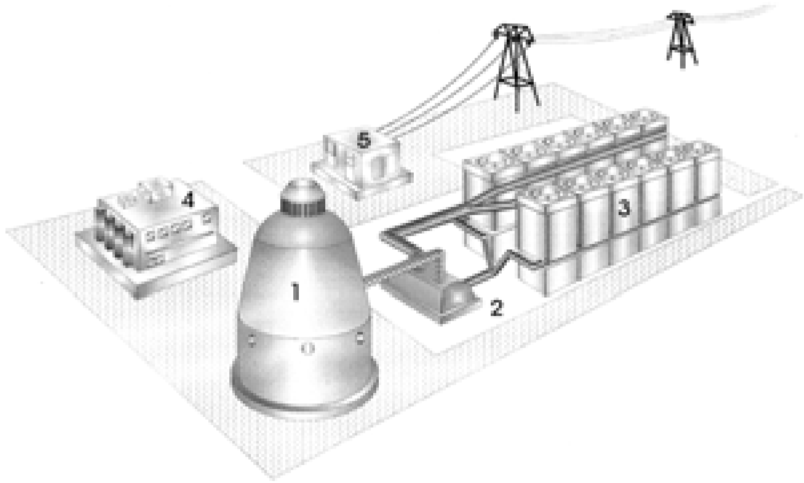
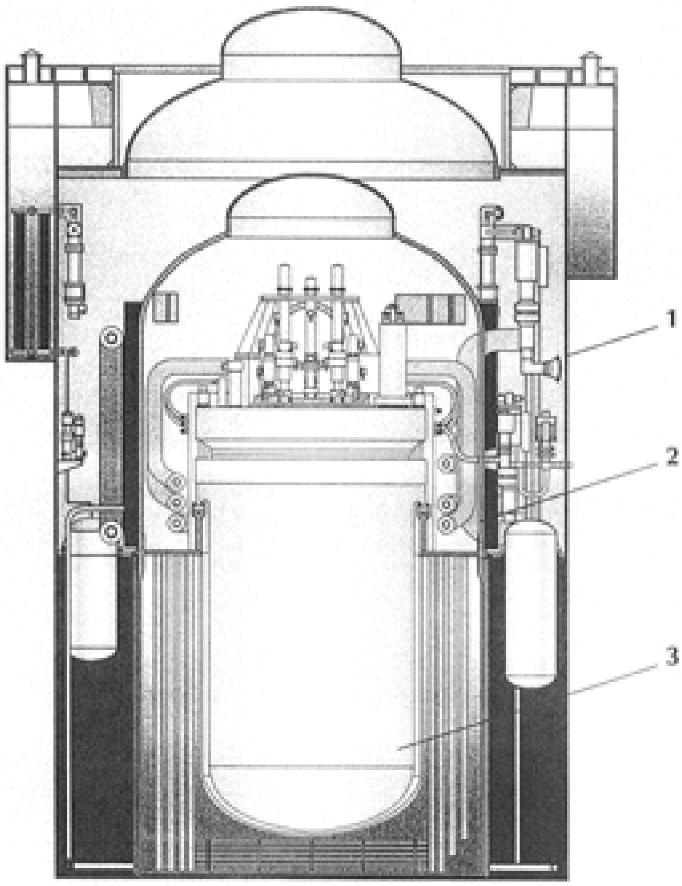


FIG. 14. Overview of UNITERM-70 NPP.



1 - Containment
2 - Guard vessel
3 - Integral reactor

FIG. 15. NIKA-70 reactor plant.

14. NUCLEAR COGENERATION PLANT WITH NIKA-70 REACTOR

The Federal State Enterprise NIKIET has developed a conceptual design of a floating NPP based on 70 MW thermal PWR of integral design with forced circulation of primary coolant and integral gas pressurizer system [38] as shown in Fig. 15. This floating NPP is developed for electric energy generation (15 MW) and may be used as power source for seawater desalination complex.

15. CONCLUSION

Small sized nuclear reactors and power plants on their basis represent a prospective class of power sources for the effective solution of several problems such as provision of isolated consumers in many regions of the world with the electric energy, district heating and/or process heat, potable water etc. The economic expediency and social importance of such power units for the provision of heat and power supply to various consumers in the extreme north and far east regions of Russia, where fuel supplies are costly and unreliable, has been comprehensively justified through a series studies. The task of development and deployment of small NPPs was included into the Special Federal Programme of Russia on "Cost-effective economics … for the period till 2010". Some sites for top-priority deployment of small NPPs have been selected. It is planned to construct a pilot small power NPP with KLT-40S reactor in Severodvinsk in the North of Russia.

For the conditions of operation in the regions of the Russian north, the use of floating (barge-mounted) small NPPs with improved marine-type PWRs is proved to be most efficient. A multi-year Russian experience in the design and operation of marine reactors provides the basis for corresponding small NPP projects.

The basic advantages of floating power units are as follows:

- Considerable reduction of the time (down to 4-5 years) and cost (by 1.4-1.6 times) of power unit construction in comparison with ground-based NPPs with an option of further decrease of floating power unit costs through standardization and mass production;
- Factory fabrication and testing of power units, i.e. their supply to a customer on a turnkey basis;
- Minimization of proliferation risk through the exclusion of any operations with fresh and spent nuclear fuel at the place of floating plant operation, or through adopting measures to prevent unauthorized access to nuclear fuel during fuel handling at the site;
- Floating NPP could be leased (rented) under "build-own-operate" conditions to any country of the world, which may make it possible to overcome the existing political and economical barriers on the way of application of nuclear power in developing countries;
- Floating power unit may be used as part of desalination complex, which can considerably increase the number of countries interested in the development of this technology;
- Floating NPP option essentially simplifies the requirements to site selection and decommissioning.

The design studies performed in Russia confirm the possibility of a considerable evolutionary perfection for both marine-type reactors and power units on their basis to meet in full the specific requirements to future NPPs with small reactors. In particular, the possibility to exclude any operations for fresh fuel handling at the site for power units of up to 110 MWe was confirmed. The possibility to develop small reactor cores capable of operation without on-site refuelling over the period of 10-12 years was proved, which would allow to combine refuelling with the repair and maintenance of floating power unit at a specialized plant.

Along with the designs based on proven reactor technologies, some innovative concepts of small reactors that meet the definition of a "nuclear battery" - power source for long-time unattended operation - are under development in the Russian Federation. Bringing such concepts to practical realization requires a substantial amount of R&D to be performed and a demonstration prototype to be created. The important tool for speeding-up the development of such concepts may be international cooperation coordinated by the IAEA, e.g. within its INPRO project.

It may be recommended that Russian designs of NPPs with small reactors SAKHA-92, ABV-3, ABV-6, RIT, VBER-150, MARS, VKR-MT, SVBR-75/100, ELENA, RUTA-70, UNITERM, and NIKA-70 are described in the new IAEA report on the status of innovative small and medium sized reactor designs. For coordinated international projects, innovative small reactor concepts based on coated particle fuel and molten salt, gas and water coolants may be recommended.

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ANNEX 7

OVERVIEW OF RRC KI PROPOSALS FOR NUCLEAR ENERGY SYSTEMS WITH SMALL AND MEDIUM SIZED REACTORS

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Abstract. The paper gives an overview of the proposals from Russian Research Centre “Kurchatov Institute” (RRC KI) for the development of nuclear energy systems that would couple nuclear power plants with small and medium sized reactors to service enterprises. The problems and R&D needs associated with such systems are analyzed. A conclusion is made that institutional changes in the organization of nuclear power production are needed to make such systems viable.

1. INTRODUCTION

The paper presents the vision of the Russian Research Centre “Kurchatov Institute” on how a system of nuclear power plants (NPPs) with small and medium sized reactors (SMRs) and their service enterprises could be built.

The complete nuclear energy system that comprises both NPPs and service enterprises is considered as an innovative project. The basic reasons behind this are that such system may provide essentially new qualities and that it requires a substantial amount of R&D to be developed and deployed.

The paper includes four sections. The first one – an introduction – describes areas where nuclear energy systems with SMRs could be applied. This description is provided on a semi-qualitative level and targets a definition of the image of possible future customers of such systems.

The second section gives a short summary of the RRC KI’s efforts in R&D for new SMR designs and systems on their basis.

The third section describes the RRC KI’s vision of the structure of future nuclear energy systems in general and systems with SMRs in particular, with a focus on the possible role and place of the latter in the overall energy system.

What could be the attractive areas for future investments in Russia?

A brief look on the advantages of certain commitments in national economy provides an understanding of some optimum ways to develop nuclear energy systems.

The major merit factors of national economy are:

- Availability of natural resources (including non-renewable: oil, gas, coal and uranium, as well as renewable ones: hydro, geo-thermal and wind);
- Large territory and well-educated and skilled labour resources;
- Stable political system.

The major demerit factors of national economy are:

- Problems of institutional development associated with young democracy;
- Severe climatic conditions, etc.

One of these demerit factors, severe climatic conditions, is actually a favourable condition for the deployment of nuclear power.

Large areas in the North, Far East and North East of Russia offer large deposits of natural ores that are not supported by adequate value of fossil resources, such oil, gas and coal [1]. However, extracting mineral resources requires energy for both industrial and residential use. Based on this, it could be concluded that potential consumers of nuclear power may be the inhabitants of small and medium settlements in the far and north east of Russia.

To meet their requirements a very sparse energy system should be built, since the locations of these settlements are sparse too. They need district heating, electricity, water purification (optionally), and process heat for technological complexes, for coal gasification and, in far future, for hydrogen production. They are ready to pay more for energy than it is being paid in the central regions of Russia, and the difference in costs per kW-hour may reach five and more.

In these regions there are no opportunities to employ large number of qualified workers on a long-term basis, and the construction and balance and commissioning period for an NPP should be very short. The same is true for NPP operation and decommissioning staff.

Based on the above considerations, user requirements for nuclear energy systems in these regions could be defined. One should note that 'user' is not identical to 'consumer' here, since the former is understood as an institution to which responsibility on making a decision for nuclear energy source implementation has been delegated. Therefore users should have some motivation to make a pro-nuclear decision (see Table 1).

TABLE 1. MOTIVATION OF DIFFERENT GROUPS OF USERS

User's group	Factor of value
Government	Power, including army, police, secret services, etc.
Regional and municipal authorities	Power, including taxes
Conservative business (long money)	Capital, property
Innovative business (venture capital)	Capital, property (to a higher degree in the form of intellectual property)
Voluntary organizations	Good will

Users also have an interest varying between different subject areas (see Table 2).

When developing a nuclear energy system as an innovative project, one should try to meet the expectations of all of the above mentioned user groups.

Talking about SMRs, the following user requirements should be mentioned:

- Simplification of control, operation and management;
- Reduced need in highly skilled personnel;
- Elimination of large volumes of construction work during deployment and decommissioning of an NPP;
- Flexible and adjustable design that meets the flexible structure of demand for various energy products;
- High level of safety and reliability, minimum adverse environmental impacts;
- High degree of proliferation resistance;
- Simplified procedures for radioactive waste management and decommissioning of an NPP.

TABLE 2. USER'S INTEREST IN DIFFERENT SUBJECT AREAS

User's group	Subject area				
	Economy	Safety	RWM*	Ecology	Non-proliferation
Government	Medium	High	High	High	Very high
Regional power bodies	High	High	Medium	Very high	Low
Conservative business (long money)	High	High	High	High	Low
Innovative business (venture capital)	Very high	Medium	High	Low	Low
Voluntary organizations	Low	High	Medium	High	Medium

* RWM is for radioactive waste management

To met these requirements, a complete nuclear energy system should be developed that will provide for:

- NPP construction and/or factory assembling and fuelling;
- Radioactive waste management and safeguards;
- Decommissioning of an NPP and site remediation;
- Reprocessing and disposition of wastes, and recycling of valuable materials, e.g. structural materials.

With these requirements in mind, several options of nuclear energy systems with small reactors were considered.

2. R&D ON SMRs AND SMR-BASED SYSTEMS

The RRC KI activities in this field include:

- Development of new designs of innovative SMRs;
- Modification of the existing SMR designs;
- Formulation of system requirements to innovative nuclear energy systems with SMRs;
- Performance of basic R&D.

RRC KI is developing a conceptual design of an innovative **M**icro coated particle fuelled **A**utonomous nuclear **R**eactor with molten **S**ault (MARS). Figure 1 illustrates the design scheme of MARS.

MARS is a pebble bed fuelled molten salt cooled autonomous nuclear reactor of small capacity, which has the following potential applications:

- Power supply to autonomous consumers in remote areas, including electricity, process heat, and district heating;
- Long-life core operation without on-site refuelling;
- Factory fabrication and fuelling;
- Potable water production (seawater desalination or water purification).

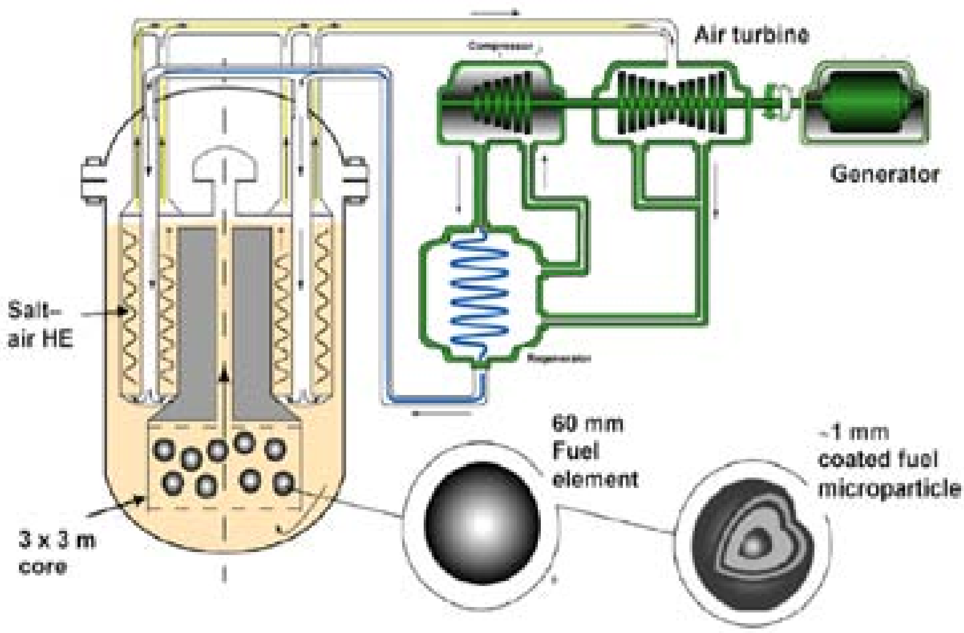

FIG. 1. Principal scheme of MARS.

The design features of MARS are outlined in Table 3.

TABLE 3. SOME DESIGN AND OPERATING CHARACTERISTICS OF MARS

Core and first circuit	
Thermal capacity, MW	16
Fuel lifetime, years	15...60
Core diameter/height, m	3 / 3
Average power density, MWt/m^3	0.75
Maximal fuel temperature, ^{0}C	1200
Coolant temperature T_{outlet}/T_{inlet}, ^{0}C	750/550
Coolant flow, kg/s	29.4
Fuel element charge, g	7.90
Fuel enrichment of ^{235}U, %	10.0
Fuel burnup, GW·day/t	98
Fast neutron fluence for a fuel lifetime, n/cm^2 fuel element	$0.53 \cdot 10^{21}$
reactor vessel	$0.33 \cdot 10^{21}$
Total height of circulating circuit, m	8.8

Secondary circuit	
Number of operating heat exchangers	3
Full number of heat exchangers (including reserve)	6
Air temperature before turbine, ^{0}C	700
Air temperature after regenerator, ^{0}C	232
Turbine efficiency, %	92
Compressor efficiency, %	88
General	
Air compression index in compressor	6
Thermal efficiency (at $T_{inlet}=0^0$C)	0.37
District heating capacity, MW	8.5
Weight of reactor unit, t	~132
Weight of GTU with electrogenerator, t	~26.4
Diameter/height of reactor unit, m	4 / 10

RRC KI also participates in the development of a project of floating NPP with KLT-40 reactor.

KLT-40 is barge-mounted NPP for multi-purpose application (see Fig. 2), including:
- Co-generation of electricity and heat for district heating;
- Potable water production.

For KLT-40, both fresh and spent fuel is placed in an on-board storage. The decommissioning procedure for a user is reduced to towing the barge to a service base.

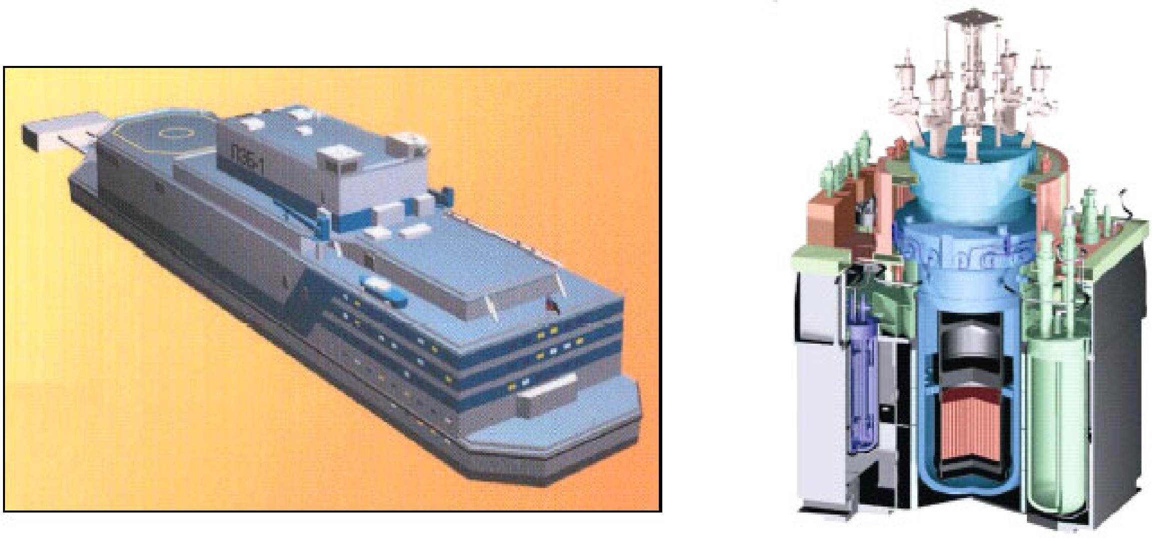

FIG. 2. Views of a NPP and a NSSS with KLT-40 reactor.

The participation of RRC KI in KLT-40 project proceeds along with the following trends:

- Studies of new types of metal–ceramic fuel to demonstrate its stability under irradiation;
- Study of non-activated (weakly activated) materials to simplify decommissioning of an NPP;
- Study of an option to reduce enrichment and to extend operation period without on-site refuelling;
- Development of a method and software to prove the absence of severe consequences in beyond design accidents (so-called “soft” mathematical models of dynamic processes).

Another SMR design developed at RRC KI is the ELENA reactor.
ELENA is a nuclear power plant with direct conversion water-cooled reactor, which supplies electricity and heat and is designed for unattended lifetime operation without on-site refuelling. The design features of ELENA ensure high reliability and safety, eliminate adverse environmental impacts, and make this NPP an attractive source of heat and the power supply for small settlements located in remote areas, including seismic and draught ones, as well as in uninhabited or underwater stations, e.g. robotized systems for investigation and extraction of ocean resources, or hydrology research laboratories.
The use of a self-adjustable water-cooled reactor coupled with thermoelectric mode of heat conversion and natural circulation of coolant makes it possible to exclude movable elements from the technological circuit of an NPP and to secure a lifetime unattended operation without on-site refuelling in the course of 25 years.

The basic characteristics of an NPP with ELENA reactor are as follows:

Capacity of heat supply system	3 MW
Electric power	up to 100 kW
Temperature of water in heat supply system	up to 90°C
Power unit dimensions:	
o Diameter	4.5 m
o Height	12-14 m
Weight (without primary and secondary coolant)	160 t
Potable water production	60 m^3/hr
Period of operation without on-site refuelling	up to 25 years

3. DEVELOPMENT OF NUCLEAR ENERGY SYSTEMS WITH SMRs AND THEIR SERVICE ENTERPRISES

Though there are multiple R&D on-going, nevertheless, even at the moment it is quite clear that no single NPP design or project alone would be able to meet all contemporary and, moreover, future consumer's needs.

There is no way to satisfy all future needs other than setting a combination of NPPs of various types, e.g. one for production of electricity, another one for district heating or process heat production and, maybe, hydrogen production etc. It is clear that such combination should include also fuel cycle enterprises. If such enterprizes are not foreseen, this may result in a negative experience, similar to what is now observed with the absence of service enterprizes to realize radioactive waste management and decommissioning strategy.

At the same time, Russian Federation has acquired a certain experience in the successful solution of a problem of development and building of service infrastructure for decommissioning of nuclear submarines. In a sense, the R&D carried out at RRC KI for the development of nuclear energy systems with SMRs and service centres substantiation of development of systems coupled NPPs with SMRs and service centres are just the reflection and implementation of a tremendous experience accumulated in other areas of atomic energy application.

The R&D started with producing an image of the future nuclear power system, and our vision of such system is given in Fig. 3.

The system considered consists of two parts. The first part is an external one. It is a network of NPPs with small and medium sized reactors having the highest level of safety and providing with different kinds of energy various technological enterprises coupled with them (power sources, built-in technological processes, etc). It is assumed that operation of such reactors does not require a large number of qualified operators.

The second part is an internal part. And it includes the nuclear installations ‘separated’ from the outside world, such as fuel manufacturing and reprocessing facilities, systems for partitioning and transmutation of hazardous radioactive nuclides, service enterprises for repair and decommissioning of NPPs, etc. All enterprises of this second part are concentrated in the so-called central repair & reloading bases (CRRB), and are operated under strict control and security measures

Creation of the abovementioned nuclear systems poses certain challenges associated with:

- The need to reduce the number of staff, as else the ratio of personnel to capacity will be inappropriate;
- The need to reduce activation of structural materials and construction elements of an NPP;
- The need to simplify decommissioning procedures.

Reduction of initial costs for systems with SMRs can be achieved by postponing the start-up of service enterprises shown in the bottom part of Fig. 3, i.e., by constructing NPPs first. The main advantage of such system is flexible application of SMRs, with corresponding risks for the clients (energy generating companies) being transferred to the supplier of nuclear technologies and services.

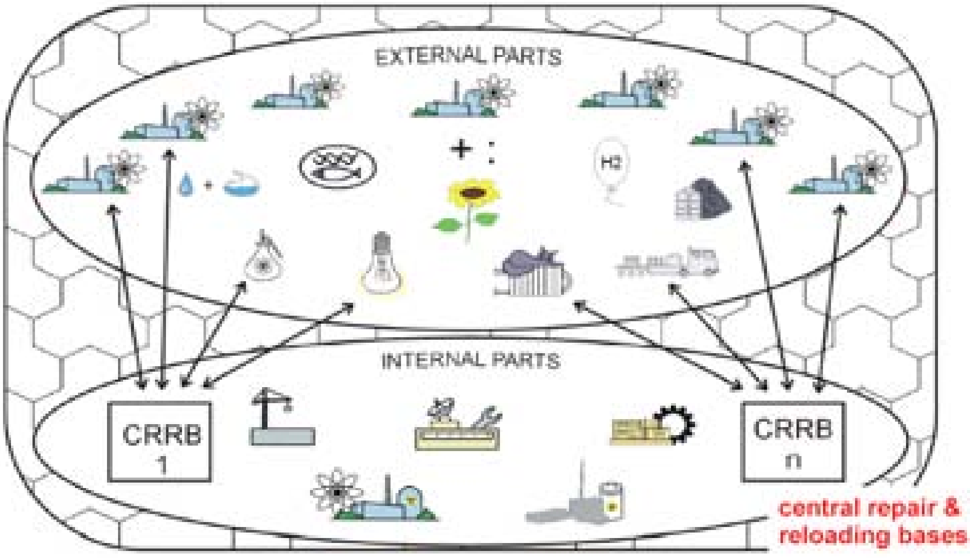

FIG. 3. Structure of the future nuclear power system.

An important challenge for nuclear power deployment is the requirement of finding a solution to the problem of radioactive wastes. In some sense, it is a natural limitation for flexibility of SMR-based nuclear energy systems, because increase in the number of NPPs will also initiate very long-term processes associated with radioactive wastes. As seen from Fig. 4, core lifetime of an SMR may be only an order of magnitude lower than the lifetime of a settlement (industrial/residential) to which this SMR is supposed to cater. However, the associated fuel cycle and decommissioning services may need to last for several times as long, while concerns with radioactive waste and spent fuel storage may protrude over a period that is several orders of magnitude longer.

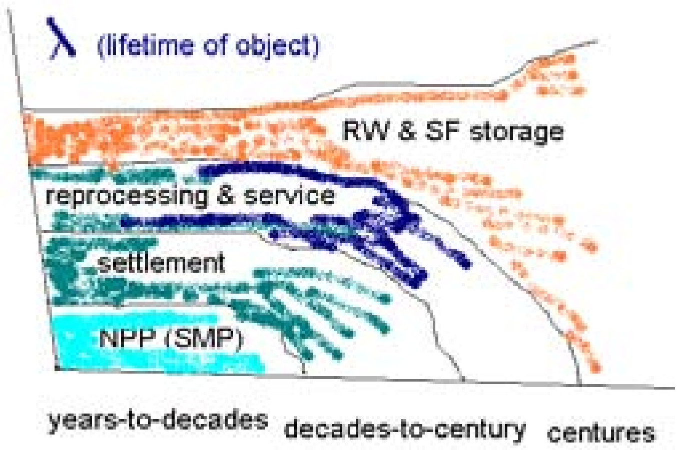

FIG. 4. Time scale of processes associated with SMR-based nuclear energy system (SMP is for small and medium power, RW is for radioactive waste, and SF is for spent fuel).

The volume of wastes produced by NPPs is not small (see Table 4) and should be taken into account when a nuclear power system is being developed.

TABLE 4.

Type of wastes	Operation, kg per MW year	Reprocessing, litre per MW year	Decommissioning, litre per MW
High level wastes	20-25	90-112	60-150
Low and intermediate level wastes	100-120	7000-9000	9000-12000

The following institutional problems were defined for systems with SMRs:

– When leasing an NPP, e.g. barge-mounted is considered, a clear definition of what is being offered for lease should be provided: is it fuel, nuclear power plant or something else;
– In defining the object of leasing, one should have in mind that, being used, nuclear fuel looses its initial property to produce heat, and that the situation may be essentially the same when leasing of a small reactor -"nuclear battery" is considered;
– Users should be guaranteed from all kinds of political risks.

Among the technology development problems for SMRs, the following should be mentioned:

– To reach the goal of flexible application, operation and management (including simplified decommissioning) it is necessary to demonstrate low level of activation in structural materials and radiation shielding of SMRs;
– Reliable proofs of negligible probability of severe accidents in SMRs should be provided;
– Evaluation of decommissioning costs for SMRs.

4. USER REQUIREMENTS TO NUCLEAR ENERGY SYSTEMS WITH SMRs

The studies performed at RRC KI make it possible to formulate the following user requirements to nuclear energy systems with SMRs:

– High level of diversity: diversity is a strong point for SMRs in energy markets. SMRs should differ in design, capacity, type of fuel and the range of products offered (electricity, low and high potential heat, co-generation, e.g. electricity and heat, potable water or hydrogen production);
– Flexibility is not a property of SMRs or SMR-based systems - it is a user requirement. Therefore, a system of NPPs with SMRs and service enterprises should support a variety of SMR designs and flexibility in product offers (electricity, heat or hydrogen);
– As decommissioning and disposal of radioactive wastes are most challenging issues for systems with SMRs, their resolution should become a principal task for the development of such systems;
– All risks associated with SMR-based nuclear energy systems should be identified and ways to insure from these risks both users and suppliers should be defined.

5. CONCLUSIONS

The Russian Research Centre "Kurchatov Institute" (RRC KI) develops a project of innovative nuclear energy system with small and medium sized reactors (SMRs). NPPs with SMRs have a number of attractive features, including simplified operation, control and management, high levels of safety and reliability, options for effective management and insurance of risks, flexible design and applications.

Systems with SMRs are considered as subsystems of the whole energy system, and they should be harmonized, first of all, by materials flows.

Within a nuclear energy system, NPPs with SMRs may be coupled with centralized service centres, which would provide the following advantages:

- Structural reliability of energy systems: they will not go down or lose revenue because of small perturbations caused by changing market conditions, including those caused by appearance of new closing technologies;
- Better harmonization of power generating and service systems, including fuel reprocessing;
- Easy adaptation of the whole energy system to consumer needs through synergetic coupling of large-power and small-power systems.

Therefore, nuclear energy systems with SMRs may introduce new quality of power generation.

Each SMR considered has a prototype in one or several military, marine or research reactors, besides being based on certain experience and technology of large sized reactors. Service systems may include refuelling of nuclear "batteries", an essentially innovative component.

Complete nuclear energy systems with SMRs and service centres have a chance to become a prototype of large-scale nuclear power systems with closed fuel cycles, the more so as full fuel cycle chain for the latter in not available yet.

Regarding the specific issues to be addressed for systems with SMRs, the recommendations of this paper are as follows:

– It is important to make a comprehensive consideration of decommissioning strategies for SMRs being developed;
– It is important to consider risk management and market opportunities for various SMRs and systems on their basis.

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ANNEX 8

SMALL PWRs WITH COATED PARTICLE FUEL FOR DISTRICT HEATING

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Abstract. It has been many times pointed out that nuclear energy is indispensable for sustainable development not only in industrialized countries but also especially in developing countries. In developing countries the population growth and economic development are progressing quite rapidly, and therefore the energy demand is higher in those countries. However, for many developing countries it would be quite difficult to deploy a domestically designed first-of-a-kind nuclear power plant, even if it is small sized. It also would be preferable to deploy plants based on proven technology. From this point of view, it is essential that industrialized countries demonstrate the reliability of a plant in advance. On the other hand, the deployment of nuclear power at large scale is not progressing well. In our study we try to define a new approach to break through this stagnation.

1. INTRODUCTION

It has been acknowledged that nuclear energy has a strong role to play, particularly if the goal is truly sustainable development. However, it is also true that deployment of nuclear energy can be said stagnant on a global scale except for a few countries. What could be done to change this situation? There have been many discussions on this in the nuclear communities around the world. Recently, there appeared some positive signs indicating brighter future for nuclear power. For example, the activities of GIF and INPRO were started and are in progress.

There are some good news for nuclear power development activities in Japan. An educational programme proposed by the Tokyo Institute of Technology has been awarded as one of the 21st Century Center of Excellence (COE) programmes in the field of innovative nuclear energy systems. The title of this program is “Innovative Nuclear Energy Systems for Sustainable Development of the World.” COE is a new programme started by the Japanese Ministry of Education, Culture, Sports, Science and Technology (MEXT) in fiscal years 2002 and 2003. Within this programme, some leading Japanese university-based research institutes were granted the status of internationally competitive research bases. A proposal approved by the COE receives a financial support for five years. This support is to be used to establish a research base that meets the highest standards known worldwide. Within the programme “Innovative Nuclear Energy Systems for Sustainable Development of the World” a survey of public opinion on nuclear energy was carried out as the first step. At a kick off meeting held in the beginning of this year some of the results were disclosed. It was expected that those questions and answers could provide some hints to define a breakthrough approach for nuclear power through analysing how the ordinary people feel about the energy problem. The outcome was as follows. Many people were concerned about greenhouse effect, environmental protection, waste reduction problem and recycling as an option, and energy saving. However less people believed that a progress in science could solve energy problems, as well as expressed their willingness to use public transport instead of private cars and to rely on natural energy. In other words, people were concerned about matters directly related to them but not about those that cannot be seen directly. Although these results were obtained in Japan, it seems that such attitude may be common all over the world. In nuclear energy community, such concept is sometimes referred to as “ NIMBY: Not In My Back Yard.” We believe that it is very important for the smooth deployment of nuclear energy on a global scale to change NIMBY to “CIMBY: Come Into My Back Yard or Construct In My Back Yard.” Changing to CIMBY can solve the problems of public acceptance. However, conventional improvements in technical aspects of power production are not sufficient to realize such a change.

Based on these considerations, we try to find a different approach for the deployment of nuclear power by proposing a plant concept that could be located in the immediate vicinity of a populated area and would provide convincing evidences of its high reliability to people living around. We plan to cooperate with the people from very early design stages by providing detailed explanations of the plant system design, pointing to how it secures establishing a very high level of safety and reliability and also showing the direct advantages people could get from a plant located in their immediate vicinity. We will also discuss the role of nuclear energy in saving the environment. We believe that a concept of a nuclear system based on small reactor without on-site refuelling could be the most suitable system to achieve our goals. As a first step, we propose for discussion a concept of a small PWR with coated particle fuel for district heating.

2. BASIC REACTOR CONCEPT

The public opinion survey has clearly indicated that the most important concern related to nuclear energy is that of radiation dosage. Radiation cannot be seen or felt directly. There is a strong belief that the dosage of radiation should be as low as possible. Although such belief may get changed in the future, it will still take a long time. Thus a CIMBY concept should meet the requirement of no radiation release under any conditions. Only when this is guaranteed there is no difference between nuclear plants and the conventional industrial plants, which are built close to the housing area.

Another issue to be solved is the problem of public acceptance. This problem cannot be solved easily by technical discussion itself. It is important that communication to the public makes it possible for all people to understand safety, reliability, design philosophy and other features of the whole nuclear power plant to their satisfaction. A very good way of getting people satisfied is to provide them rigid proofs of an excellent operating experience regarding the technology to be used. From this point of view, we believe that the operating experience of light water reactors around the world is a good example. They have shown excellent performance during recent decades. We should not forget the fact that these good operating records have been supported by a large amount of research and development and investment in the development of technology for fuel, structural materials and plant maintenance as accomplished by many governments and industries around the world. We should make maximum use of these precious data for the future success of nuclear power.

Based on such consideration we have proposed a reactor concept, which is a small PWR based on coated particle fuel packed within a conventional Zircaloy cladding. It is well known that coated particle fuel has an excellent capability of fission product confinement up to about 2000 deg. C [1]. The reason we use conventional cladding is to make advantage of the long operating experience of PWR fuel rods and to avoid chances for the particles to contact air or oxygen directly during accidents. This will help to assure elimination of radiation release. The particles are loaded in a fuel rod with graphite to make a composite and provide additional neutron moderation. The graphite also contributes to good thermal conductivity of fuel, thus keeping the fuel temperature lower than in conventional ceramic fuel pellets.

As a small reactor is economically handicapped, we try to design the system as simple as possible. We propose to eliminate liquid boron system and control reactivity by B_4C basedburnable poison. For a small reactor it is also favourable to ensure a long-term operation without refuelling. We try to design a reactor with core lifetime of 10 years under the initial fuel enrichment of 5 % (weight).

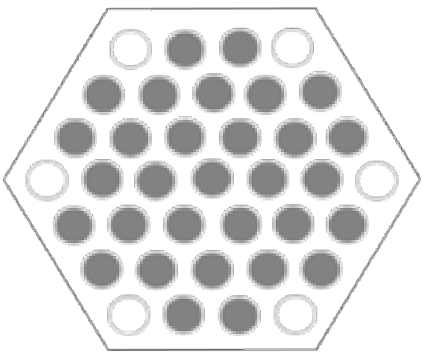
3. PRELIMINARY DESIGN

3.1. Summary of core design characteristics

Table 1 presents the summarized major design characteristics of reactor core. As previously mentioned, the proposed reactor system uses HTGR type fuel, i.e., TRISO coated particles packed in graphite matrix. TRISO fuel consists of a UO_2 sphere covered with 4 coating layers, porous PyC, dense PyC, SiC, and dense PyC once again. It is well proven that all these layers play a significant role in confining fission products at high temperatures and up to high fuel burn-ups. Moreover, SiC layer is corrosion resistant in water and steam at high temperatures and therefore protects its interior layers and fuel kernel in a PWR type environment. The volume ratio of C to UO_2 is usually 9.0 for HTGR fuel. For this fuel composition, a study has been conducted that confirmed the feasibility of a PWR with HTGR type fuel in zircaloy-4 [2,3].cladding. It was concluded that, by adding Gd_2O_3 to fuel compacts, the excess reactivity could be suppressed and flattened for 4 years at 70% capacity factor, and therefore control rod programming could be essentially simplified. However, in this study we doubled the volume of UO_2, i.e., assumed that UO_2: C = 1:4 in order to achieve longer operation period. Fuel rods have standard PWR type zircaloy-4 claddings. Compatibility of graphite and zircaloy-4 is quite good, and fuel rod and cladding can operate together retaining their chemical stability.

An assembly proposed could accommodate 37 fuel rods within a hexagonal tight lattice, which makes the size of the core as small as possible. For this preliminary design we employed two types of fuel assemblies: the so-called GT and BP assemblies, see Fig. 1. The former have control rod Guide Thimbles at each corner and contain 31 conventional fuel rods in the remaining space. The latter have 18 fuel rods that contain B_4C in their graphite matrix and are placed along the periphery, and 19 conventional fuel rods. Fuel rod pitch was selected to be approximately 3.3 cm in order to obtain highest possible values of average discharge burn-up.

In total, 85 such assemblies are loaded in the approximately 2 m-diameter core with light water reflector around it, see Fig. 2. The design thermal output of the reactor is 50 MW. Therefore, the average linear power density is only half of that in typical PWR. Light water, which acts both as coolant and moderator, is circulated up through the core and its average temperature is 250^{o}C.



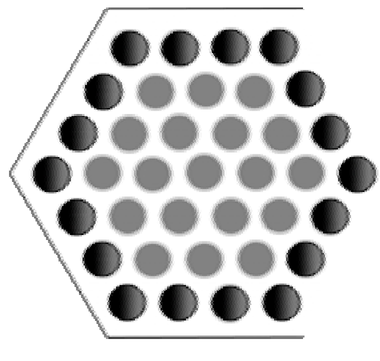


FIG. 1. Cross-section view of GT (left) and BP fuel assemblies.

TABLE 1. MAJOR CHARACTERISTICS OF THE PRELIMINARY DESIGN

Thermal output	50 MW
Fuel type	TRISO (UO_2 kernel)
Enrichment	5 wt%
Coolant and moderator	Light water
Average primary coolant temperature	250°C
Type of circulation	Forced flow
Linear power density	87.2 W/cm
Heavy metal load	5.2 t
Fuel rod pitch	~ 34 mm
Fuel rod diameter	~ 29 mm
Cladding material	zircaloy-4
Guide thimble material	SUS316 stainless steel
Lattice type	Hexagonal
Number of fuel rods per assembly	37 / 31 (GT)
Number of assemblies	85
Cladding material	zircaloy-4
Number of GT assemblies	31
Equivalent core diameter	~ 2.0 m
Effective core height	~ 1.8 m

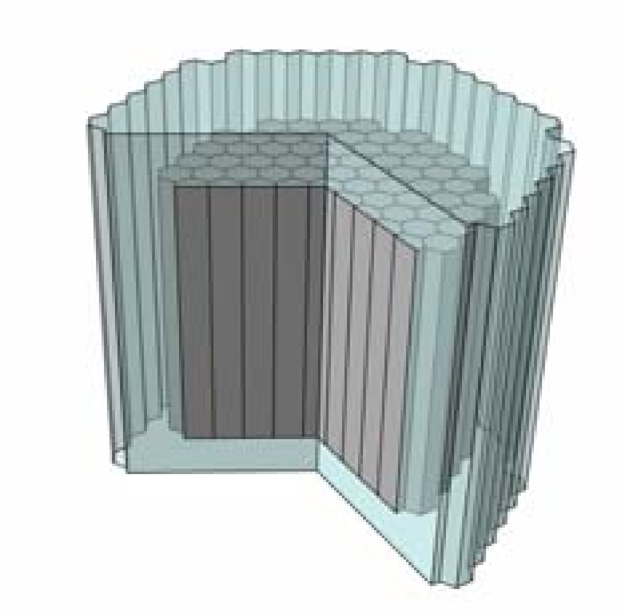


FIG. 2. The proposed core with 85 fuel assemblies.

3.2. Analysis of fuel burn-up characteristics

The code used for calculations was SRAC95 with JENDL3.2 [4].data library, which is a general-purpose reactor analysis code system developed by the Japan Atomic Energy Institute (JAERI). There are several modules integrated into the system. We have selected SRAC-PIJ for unit cell burn-up calculation and ASMBURN for assembly burn-up calculation, both of which are collision probability calculation modules. Also, COREBN, which is a 3-D diffusion calculation module, has been used for full-core calculations.

According to our analyses, it is necessary to load 31 GT assemblies to gain satisfactory shutdown margin under cold zero power condition. As for BP assemblies, 54 of them with 5 different quantities of burnable poison (Fig. 2) are required to flatten the excess reactivity throughout a cycle (Fig. 3). It seems that the reactivity control system could be simplified with such a loading pattern. The selected core configuration can achieve average discharge fuel burn-up of 25400 MWd/t and is capable of operating for 7.3 effective full power years (EFPY) without reloading and reshuffling of fuel. Also calculated were the moderator temperature coefficient, Doppler reactivity coefficient, and void reactivity coefficient at BOL and EOL. Our analysis confirmed that all of them are negative, see Table 2.

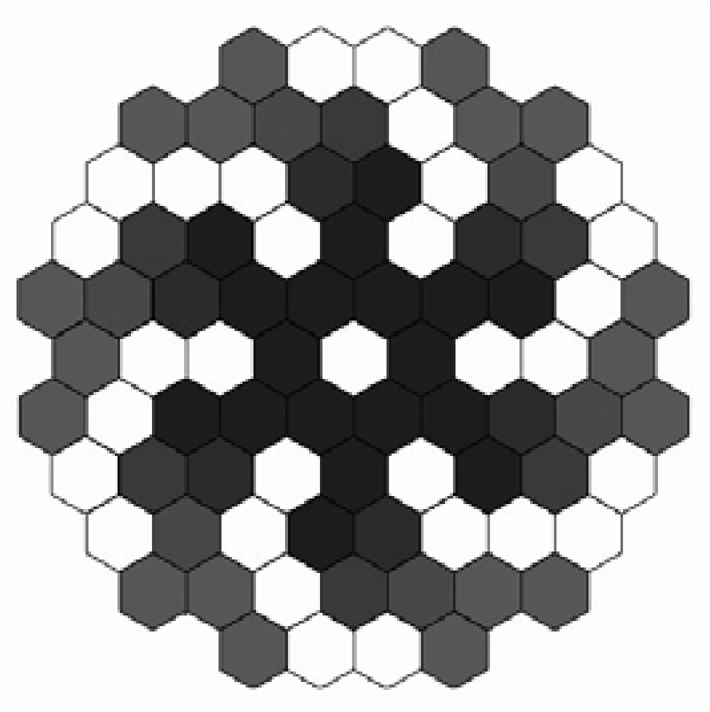


FIG. 2. Cross-section view of the core with 54 BP and 31 GT assemblies. BP and GT assemblies are in grey and white respectively.

TABLE 2. REACTIVITY COEFFICIENTS AT BOL AND EOL

	BOL	EOL
Moderator temperature coefficient [pcm/K]	-32.4	-26.2
Void reactivity coefficient [pcm/%]	-160	-151
Doppler coefficient [pcm/%]	-3.4	-7.3

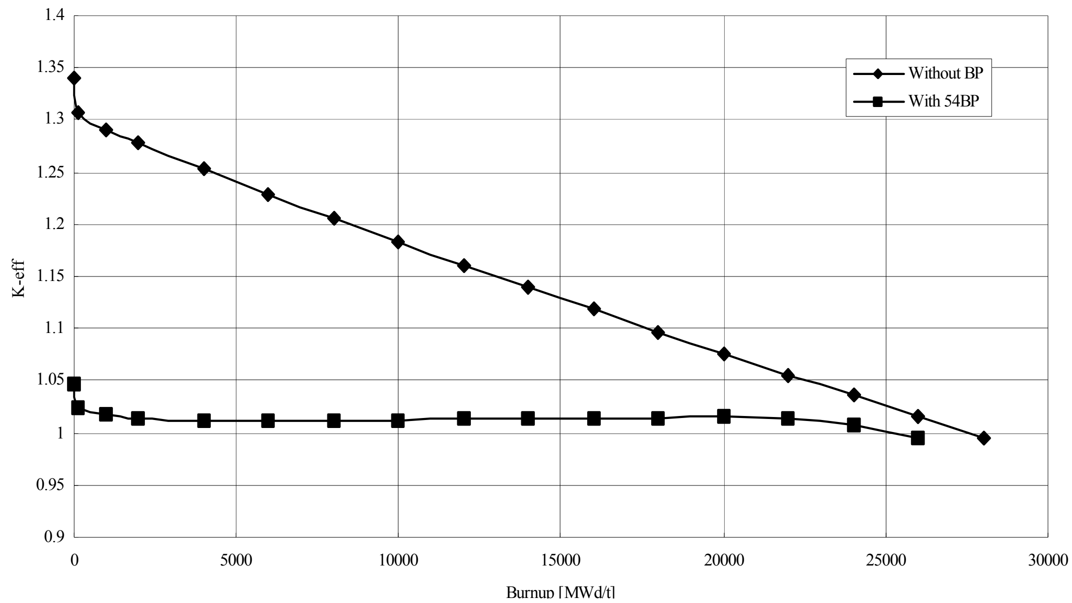


FIG. 3. Burn-up characteristics of the core with and without BP assemblies.

3.3. Safety features
The proposed reactor system has several unique safety features. One of them is that the average linear power density is approximately half of that of a typical PWR. Also, the operation temperature does not have to be so high as in conventional power plants because the proposed reactor system is aimed at district heating. Another feature is that the graphite matrix reduces the temperature rise in case of accidents because of its large heat capacity. Additionally, as introduced previously, the perfect fission product confinement capability of coated particle fuel at high temperatures should be highlighted. All these features contribute to making the thermal and safety margin of the system larger, which also may justify an extensive use of passive safety systems.

3.4. Fuel cycle planning
The reactor vessel and core would be prefabricated and shipped to the site as a single module. After operating for a designed period, the module would be replaced with a new one and cooled and stored at the site in a way that would make it possible to establish a high degree of proliferation resistance and physical protection, for example, through continuous monitoring of spent nuclear fuel. Spent fuel in the module is expected to be reprocessed when the economical reprocessing method for this type of fuel becomes available.

4. CONCLUSION

While a consensus has been reached worldwide that nuclear energy is an essential option to meet the increasing energy demand without adverse environmental impacts, it appears that there is a need of finding new, breakthrough approaches to overcome the current stagnation of global nuclear power deployment. Based on the results of public opinion survey recently carried in Japan, changing from " NIMBY: Not In My Back Yard" to "CIMBY: Come Into My Back Yard or Construct In My Back Yard" in public appreciation of nuclear power was found important to secure smooth further deployment of nuclear power plants. In order to make such a change within a short term, full advantage of the existing reliable technologies and data should be taken. With these considerations in mind, we have conducted a feasibility study to develop a truly acceptable nuclear reactor system for district heating based on light water reactor technology combined with HTGR fuel concept. We plan to select a small village with the economy at an appropriate scale and convene meetings to figure out how to establish high degree of people's confidence in safety and reliability of nuclear power plants as well as to collect a wide range of information that would be further on incorporated into the project. We will continue our efforts to complete conceptual design of the proposed nuclear plant for district heating.

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ANNEX 9

PASSIVE SAFETY FEATURES OF INDIAN INNOVATIVE NUCLEAR REACTORS

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Abstract: Nuclear fuel resource available in India in plentiful quantities to sustain a large power programme is thorium. At present, activities related to design and development of Avanced Heavy Water Reactor (AHWR) and Compact High Temperature (CHTR) reactor are underway for utilization of thorium. The R&D work related to these reactors is being carried out in BARC. The 300 MWe AHWR is a direct-cycle, boiling-light-water-cooled, heavy water moderated, vertical-pressure-tube-type reactor with natural circulation as mode of heat removal from core under all conditions. AHWR uses naturally available thorium as its main fuel resource, converts it into fissile ^{233}U, which is burnt in-situ to generate energy. Main physics design objectives are maximization of power from Th/^{233}U, negative void coefficient of reactivity, minimization of initial inventory and consumption of plutonium, self-sustaining characteristic in ^{233}U and high discharge burn-up with low excess reactivity. In addition, the AHWR incorporates several passive safety features. These include core heat removal through natural circulation of the coolant in the main heat transport system, direct injection of Emergency Core Cooling System (ECCS) water in fuel, passive systems for containment cooling and isolation and availability of large inventory of water in overhead Gravity Driven Water Pool (GDWP) for continuous long-term removal of decay heat from the core. The Compact High Temperature Reactor (CHTR), a technology demonstration reactor, the design of which is underway at BARC, also uses thorium-based fuel. The coolant used in the CHTR is liquid metal lead-bismuth eutectic. This reactor also incorporates several passive systems like removal of core heat by natural circulation of liquid metal coolant in the main heat transport circuit, passive regulation and shut down systems. The reactor is also able to remove heat passively by way of conduction in the reactor block and by radiation and natural convection from the outer surface of the reactor during loss of heat sink.

1. INTRODUCTION

Thorium is the nuclear fuel resource available in India in plentiful quantities to sustain a large power programme. At present, the design and development of Advanced Heavy Water Reactor (AHWR) and Compact High Temperature Reactor (CHTR) are underway for utilization of thorium. Incorporation of simplified and passive systems is one of the features of these reactors. The R&D work related to these reactors is being carried out in BARC.

The 300 MWe Advanced Heavy Water Reactor (AHWR) [1] is a direct-cycle, boiling-light-water-cooled, heavy water moderated, vertical-pressure-tube-type reactor with natural circulation as mode of heat removal from core under all conditions. The general arrangement of AHWR is depicted in Fig. 1. AHWR uses naturally available thorium as its main fuel resource, converts it into fissile ^{233}U, which is burnt in-situ to generate energy. Main physics design objectives are maximization of power from Th/^{233}U, negative void coefficient of reactivity, minimization of initial inventory and consumption of plutonium, self-sustaining characteristic in ^{233}U and high discharge burn-up with low excess reactivity. A composite cluster contains both (Th, Pu) MOX and (Th, ^{233}U) MOX fuel pins. The fuel assembly is suspended from the top in the coolant channel of the reactor. The assembly consists of a single long fuel cluster and two shield sub-assemblies. The cluster has 54 fuel pins arranged in three concentric rings: 12 pins in the inner ring, 18 pins in the intermediate ring and 24 pins in the outer ring. These rings are around a central rod containing burnable absorber dysprosium as ZrO2-Dy_2O_3. The 24 pins in the outer ring have (Th-Pu)O_2 as fuel pins. The inner and intermediate rings have (Th-^{233}U)O_2 fuel cluster as fuel.

Apart from thorium utilization and establishing a slightly negative void coefficient of reactivity, the AHWR incorporates several passive safety features, which include core heat removal through natural circulation of the coolant in the main heat transport system.

The Compact High Temperature Reactor (CHTR) of 100 kWth being designed at BARC also uses thorium-based fuel. Figure 2 shows the schematic of the reactor. The TRISO type coated fuel particles having kernel of $(Th\text{-}^{233}U)C_2$ is used. The reactor mainly consists of fuel tubes centrally located in hexagonal beryllia blocks. Layers of beryllia and graphite reflectors surround these blocks. The fuel tubes are made of graphite and have number of axially drilled holes in which the fuel particles are packed. Figure 3 shows the cross sectional layout of CHTR. The coolant used in the CHTR is liquid metal lead-bismuth eutectic. The main coolant-circulating loop comprises fuel tubes, downcomers and top and bottom plenums. The fuel transfers the energy to the coolant flowing upward inside the fuel tubes due to natural circulation. The coolant rejects heat in the top plenum and flows down the downcomers to the bottom plenum where it again enters the fuel tubes. The core is contained in a metallic vessel. Gas gaps are provided around the reactor block to avoid heat loss radially from the reactor block.

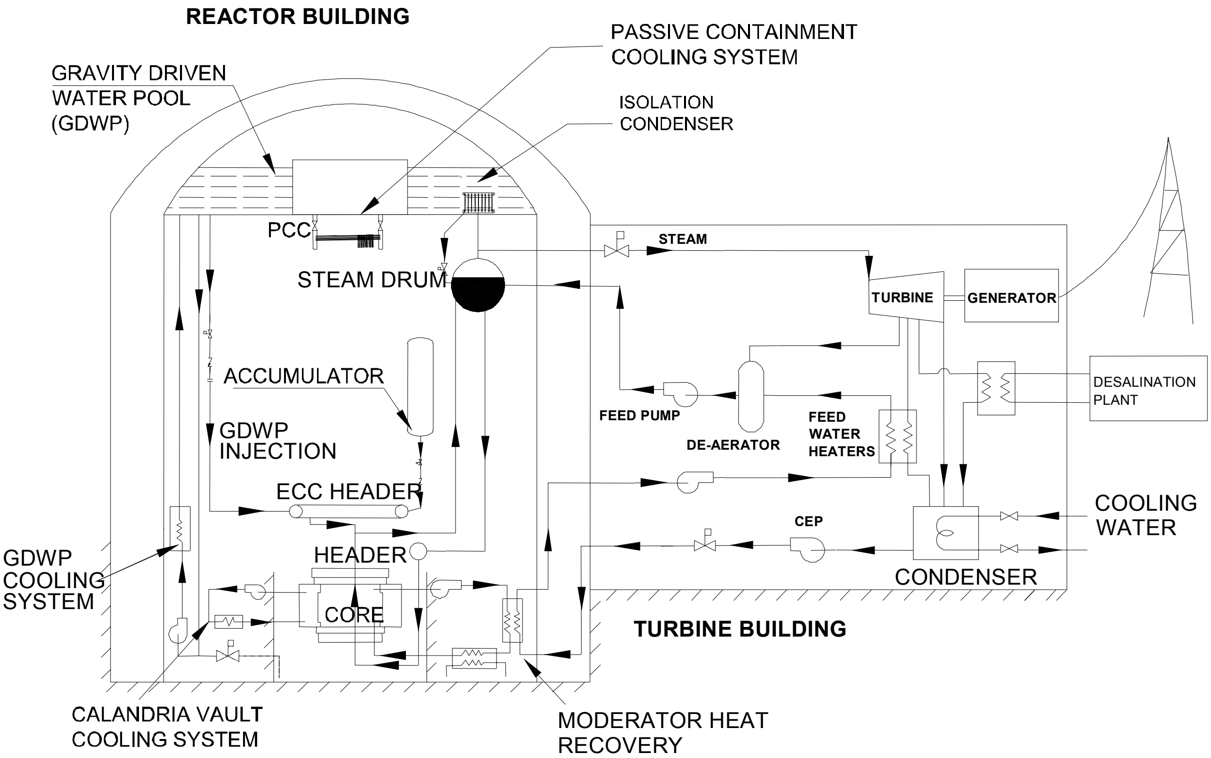

FIG. 1. General arrangement of AHWR.

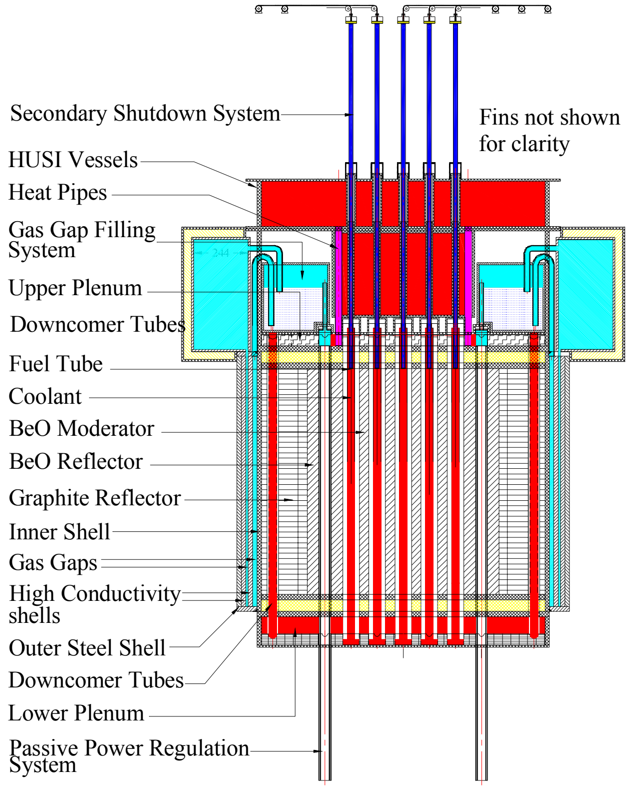

FIG. 2 Schematic of the compact high temperature reactor.

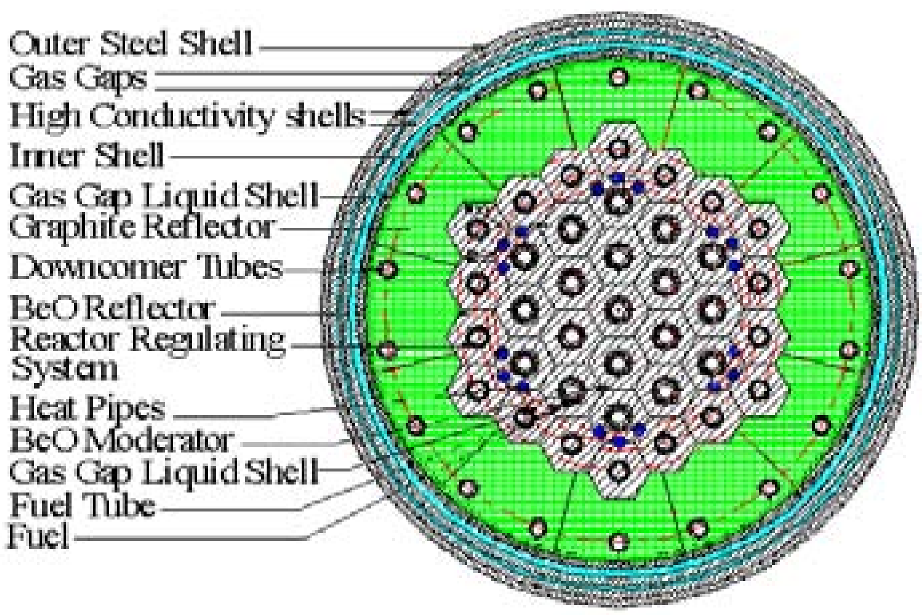

FIG 3 CHTR cross sectional layout.

Several inherent and passive safety features are incorporated in compact high temperature reactor. Due to negative temperature coefficient of reactivity, the power of the reactor comes down without necessitating any external control in case of increase in core temperature. The reactor also adopts passive systems like removal of core heat by natural circulation of liquid metal coolant in the main heat transport circuit, passive regulation and shut down systems. The reactor is also able to remove heat passively by way of conduction in the reactor block and by radiation and natural convection from the outer surface of the reactor during loss of heat sink. The paper deals with the details of passive systems incorporated in the AHWR and CHTR and the analysis performed for these systems.

2. PASSIVE SAFETY FEATURES OF AHWR

Several safety features have been incorporated in the AHWR design. The evaluation of these features is being carried out as a part of R&D work. These include the following:

1. Heat removal through thermosyphon driven natural circulation under both normal operation and hot shut down condition;
2. Direct injection of ECCS water in fuel;
3. Passive containment cooling system;
4. Passive containment isolation system;
5. Availability of large inventory of water in overhead Gravity Driven Water Pool (GDWP) to facilitate removal of long-term decay heat generated in the core of AHWR during design basis accident.

2.1 Natural circulation of primary coolant

The Main Heat Transport (MHT) system is based on natural circulation as design mode of heat removal by boiling light water. The driving force is provided by the density difference in the hot and cold leg of the MHT loop. The MHT loop, shown in Fig. 4, consists of a ring header from which 452 feeders branch out to join an equal number of channels consisting fuel clusters. The heat generated in the fuel due to fission reaction is transferred to the circulating coolant. The resulting two-phase coolant is transported to four steam drums by 452 risers leaving the core. The steam water separation is gravity affected (without mechanical separators) in the four horizontal steam drums. The separated steam flows to the turbine and the water flows back to the header through the downcomers, four each from the drum. Subcooled feed water is added to the steam drums for continued circulation. At the rated power the steam pressure is maintained at 7 MPa. The inlet sub-cooling, core average exit quality and coolant flow rate are 26 K, 18.2% and 2237 kg/s respectively.

As discussed above, during normal reactor operation, full reactor power is removed by natural circulation. The necessary flow rate is achieved by locating the steam drum at suitable height above the centre of the core. Figure 5 shows variation in coolant flow rate, void fraction and quality with power for design configuration of the reactor.

By eliminating nuclear grade main circulating pumps, their prime movers, associated valves, instrumentation, power supply and control system, the plant is made simpler and easier to maintain as compared to the options involving forced circulation in the main heat transport

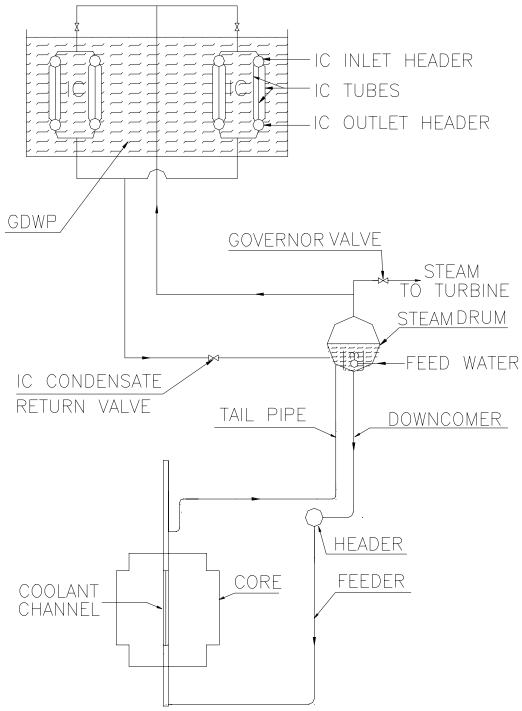

FIG. 4. Schematic of main heat transport and passive decay heat removal system of AHWR.

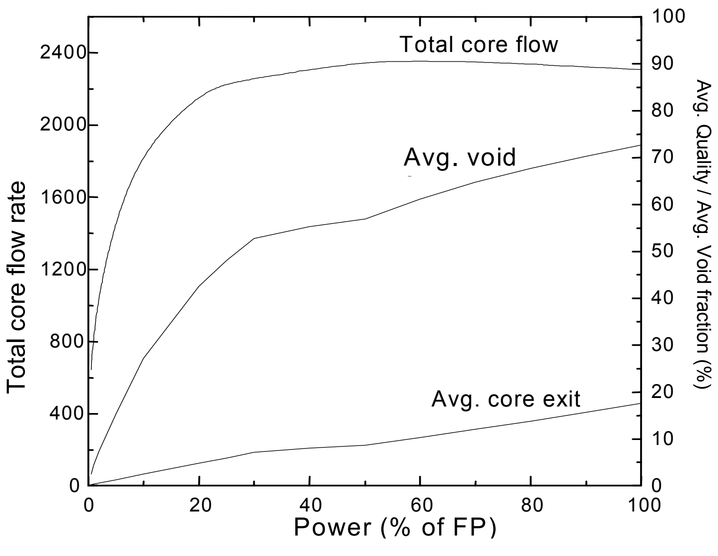

FIG. 5. Variation of total core flow, average void fraction and average core exit quality with power.

circuit. The above factors also lead to considerable enhancement of system safety and reliability since pump related transients have been removed. An integral test loop to confirm the steady state and transient analysis has been designed as R&D activity. The experimental program will establish the suitability of loop height and thermal hydraulic stability of the main heat transport system.

2.2 Core decay heat removal system

During normal reactor shut down condition core decay heat is removed by passive means by utilizing Isolation Condensers (ICs) immersed in Gravity Driven Water Pool (GDWP) located above the steam drum. Core decay heat, in the form of enthalpy of steam enters the IC tube bundles through natural circulation. The steam condenses inside the tubes and heat up the surrounding pool water. The condensate returns by gravity to the core. The water inventory in the GDWP is adequate to cool the core for more than 3 days without any operator intervention and without boiling of GDWP water. Figure 4 depicts the core decay heat removal system comprising isolation condensers. A separate GDWP cooling system is provided to cool the GDWP inventory in case the temperature of GDWP inventory rises above a set value. An active shut down cooling system is also provided to remove the core decay heat in case ICs are not available.

2.3 Emergency core cooling system

During Loss Of Coolant Accident (LOCA) emergency coolant injection is provided by passive means to keep the core flooded so as to prevent overheating of the fuel. The emergency core cooling system (ECCS) is designed to fulfil two objectives. One is to provide large amount of cold water directly into the core in the early stage of LOCA and then a relatively small amount of cold water for longer time to quench the core. This objective is achieved through use of a passive fluidic control device. Second is to provide water through GDWP to cool the core for more than 3 days.

Long-term core cooling is achieved by active means by pumping water from reactor cavity to the core through heat exchangers.

The ECCS accumulators and GDWP are connected to the PHT system by rupture disks, check valves and isolation valves kept in series. During reactor start-up, closing the isolation valves isolates accumulators and GDWP. When the PHT system pressure reaches the operating pressure level, these isolation valves are opened. The nitrogen pressure in the accumulators is always maintained at 5 MPa to keep the system in a state of readiness. Following a postulated LOCA, when the PHT system pressure falls below 5 MPa, the rupture disk opens out allowing cold water from accumulators to flow into the core. When accumulators get exhausted, low water level signal from accumulators results in closure of isolation valves and water from accumulator stops flowing into core. At this stage, water from GDWP starts flowing into the core by gravity. Through an optimum positioning of the discharge nozzles, the GDWP based ECCS flow is closely matched to the requirement for core decay heat removal, enabling an extended duration of availability of ECCS flow for more than three days. Figure 6 depicts the emergency core cooling system of AHWR.

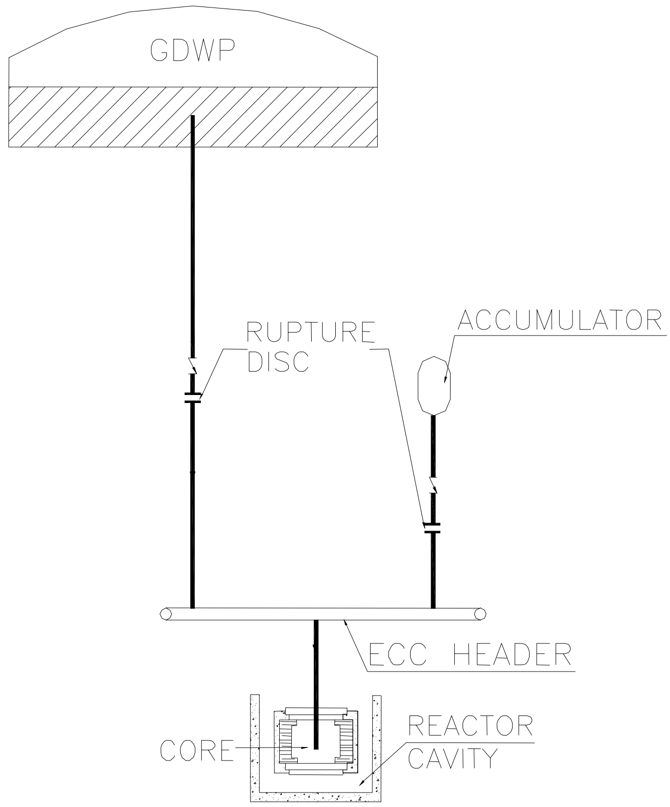

FIG. 6. Emergency core cooling system of AHWR.

Water from the reactor cavity (which is filled up with hot water after spillage from the ruptured pipe and water from accumulators and GDWP after cooling the core) is pumped back into the core through heat exchangers for long-term recirculation. This heat is transferred, in the heat exchangers, to the process water, which in turn dissipates its heat to the ultimate heat sink.

2.4 Passive containment cooling system

Containment is a key component of the mitigation part of the defence in depth philosophy, since it is the last barrier designed to prevent large radioactive release to the environment. In advanced heavy water reactor (AHWR), a passive containment cooling system (PCCS) is envisaged which can remove long-term heat from containment following Loss of Coolant Accident (LOCA). Immediately following LOCA, steam released is condensed in vapour suppression pool. For subsequent long term cooling PCCS is provided. PCCS as by definition is able to carry out its mission with no reliance on external source of energy and operates only on the basis of fundamental physical laws, such as gravity or differential pressure.

In the AHWR, a large pool of water is provided at high elevation near the top of the containment. This pool named Gravity Driven Water Pool acts as the heat sink for a number of passive heat removal systems including the PCCS. The passive external condensers (PEC) of the PCCS are connected to the pool as shown in Fig. 7. The containment steam condenses on the outer surface of tubes of PEC. The water inside the tubes takes up the heat from air/vapour mixture and gets heated up. Due to the heating up of water the natural circulation of water from the pool to PEC and from PEC to pool is established.

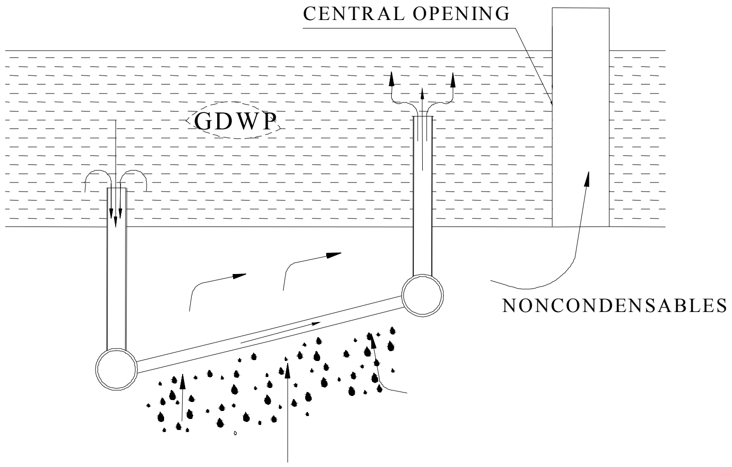

FIG. 7. Schematic of passive containment cooling system.

One important aspect of PCCS functioning is the potential degradation in heat transfer on PEC outer surface due to the presence of non-condensable gases in the containment. The presence of non-condensable (NC) gases in vapour can greatly inhibit the condensation process. Extensive R&D work is in progress to address this issue. A theoretical model to determine the heat transfer coefficient in presence of non-condensable gas has been developed. The heat transfer coefficients estimated by the model have been compared with the data available in the literature. Figure 8 shows the heat transfer coefficient as function of air mass fraction. Another aspect of PCCS functioning is the blockage of passive external condenser by non-condensable gas due to the stratification of non-condensable gas/vapour in the containment. In case of AHWR, the non-condensable gas is likely to escape through the central opening provided in the GDWP to the dome region. Experiments are planned to confirm this.

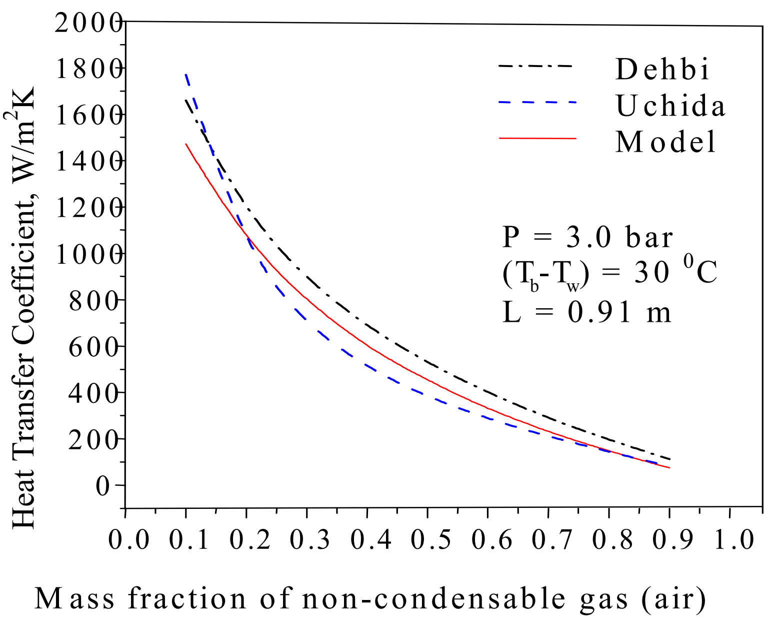

FIG. 8. Variation of heat transfer coefficient with non condensable mass fraction for stagnant mixture.

2.5 Passive containment isolation system

For containment isolation, in addition to the normal inlet and outlet ventilation dampers, a passive system has been provided in the AHWR. The schematic of the passive isolation system of AHWR is given in the Fig. 9. The reactor building air supply and exhaust ducts are shaped in the form of U bends of sufficient height as shown in Fig. 9. The reactor has double containment system, including primary and secondary containments. Between the two containments, a negative pressure with reference to atmosphere is maintained to ensure that there is no release to atmosphere under all conditions. The primary containment envelops the high enthalpy and low enthalpy zones designated as volume V1 and volume V2 respectively. These two volumes are communicated via vent shaft submerged in the GDWP. Under postulated LOCA, the steam releases into the volume V1. The steam mixes with the air present in the volume V1 and the mixture enters the vent shaft submerged in the GDWP. The steam gets condensed in the GDWP while air gets accumulated in the volume V2. Due to this, the volumes V1 and V2 get pressurized to different pressures. From the safety point of view, it is important to note that the pressurization of the two volumes is accompanied by release of radioactivity from V1 and V2. Hence under accidental conditions, it is of paramount importance to isolate this volume from atmosphere in a minimum possible time. This is accomplished by making use of the differential pressure rise in both volumes through the proposed isolation scheme.

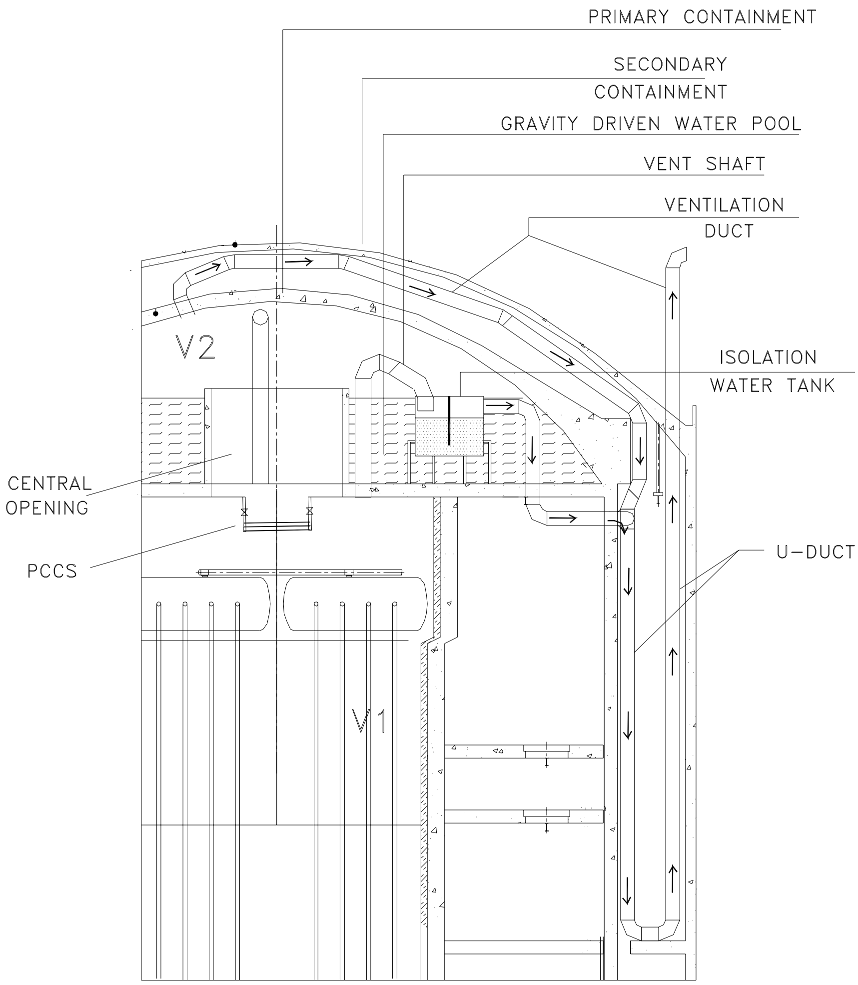

FIG. 9. Passive containment isolation system of AHWR.

The scheme consists of an isolation water tank comprising two compartments, one is in communication with volume V1 through a vent shaft while other is in communication with volume V2 via the top portion of ventilation duct. A vertical baffle plate, running from the top of the tank, separates the two compartments. The bottom portion of the tank allows the two compartments to be in communication. It may be noted from Fig. 9 that volume V2 is ventilated to atmosphere through a 'U-shaped' duct, which has a branched connection to isolation water tank outlet. Therefore, under any operating conditions, one compartment of the tank would experience the pressure of V1 volume while the other compartment of the tank would experience the pressure of V2 volume. Hence, at steady state, the water levels in two compartments would differ by hydrostatic differential pressure head between the two volumes. In event of volume V1 pressurizing to certain pressure beyond the V2 pressure, the water level in other compartment of the tank rises to spill the water into the U-shaped duct. Thus, securing a water seal at the base of U-shaped duct ensures isolation of volumes V1 and V2 from atmosphere. It is required that the seal is formed in a minimum possible time, typically of the order of a few seconds, to ensure fast isolation. In the light of this requirement, it is essential to verify the response of the system and design the system accordingly.

Towards this objective, a theoretical model has been developed and based on this model, the response of the system has been estimated using the indigenously developed computer code CAPCIS (Code for Analysis of Passive Containment Isolation System). The numerical analysis performed takes into account the pressure transient because of the postulated initiating event (PIE) and the solution includes determination of levels in the two compartments of the isolation tank and the cumulative volumetric spill into the ventilation duct at any point of time during the transient. A case of a 200% break in inlet header is considered for the analysis and the pressure transient for the considered PIE is numerically simulated as shown in Fig. 10. Figure 11 shows the volumetric spillover during the entire transient.

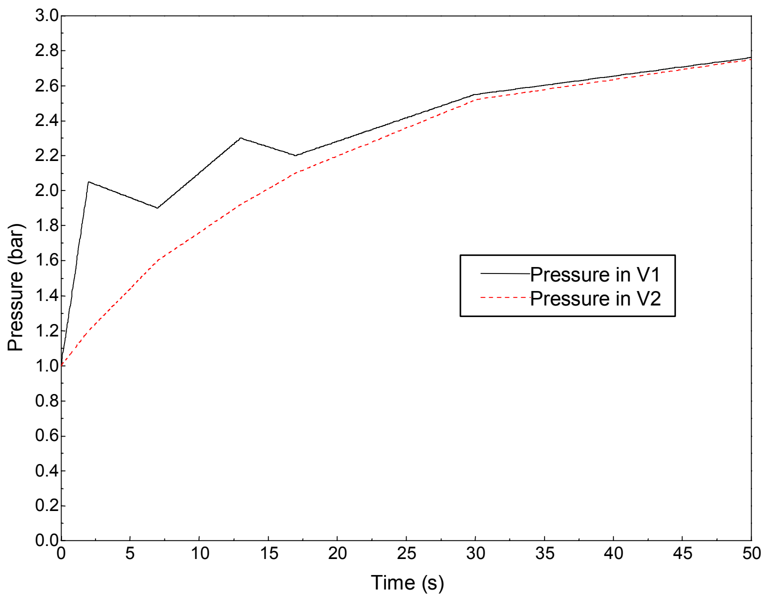

FIG. 10. Simulated containment pressure transient.

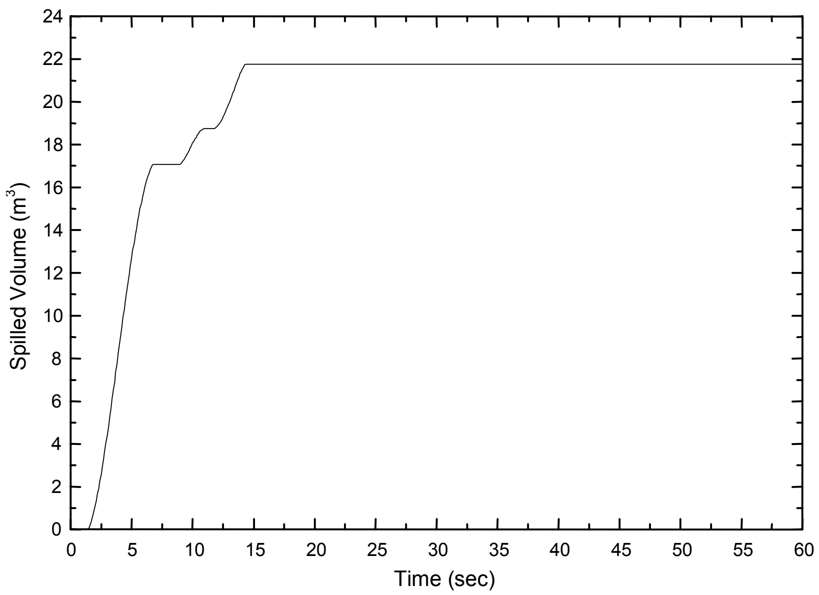

FIG. 11. Cumulative spilled volume with time.

3. PASSIVE SAFETY FEATURES FOR CHTR

A number of inherent and passive safety features have been incorporated in the CHTR. The CHTR is being designed on the following guidelines:

1 Use of thorium-based fuels with low fissile inventory and maintaining the negative fuel temperature coefficient of reactivity throughout the reactor operation;
2 Passive core heat removal by natural circulation of liquid heavy metal coolant;
3 Passive rejection of entire heat to the atmosphere under accident conditions;
4 Passive power regulation and shutdown systems.

3.1 Natural circulation of heavy liquid metal coolant

The reactor operates at 100 kWth and the lead-bismuth eutectic alloy coolant flowing in the main heat transport system by natural circulation removes the heat generated in the fuel. Lead and lead alloys have extremely high boiling point (~1670-1750^{o}C) at atmospheric pressure. This facilitates an ambient pressure primary system, which is a safety feature of liquid metal cooled reactors. The main coolant-circulating loop comprises fuel tubes, downcomers and top and bottom plenums. The coolant transfers the energy to the coolant flowing upward inside the fuel tubes due to natural circulation. It enters the core at 1173 K and leaves at the 1273 K. The active heat generation length in the reactor is 700 mm. The buoyancy head developed in the coolant loop is adequate to maintain the required flow rate for normal power level. The heat from the upper plenum is transported to another plenum using the heat pipes, which act passively without significant drop in temperature. An experimental program to investigate issues related to natural circulation and corrosion of structural material, etc. has been taken up in India. A liquid metal coolant loop has been designed for this purpose.

3.2 Heat removal during loss of heat sink

The CHTR is capable of rejecting heat to the atmosphere by passive means at neutronically limited peak power level, without fuel damage. Two gas gaps are provided around the reactor block to avoid significant heat loss radially during the normal operating conditions. During the postulated events like loss of heat sink or loss of flow, these gaps get filled up by high conducting liquid metal from the tanks kept adjacent to the top plenum. The gas headers provided in the upper plenum of the reactor get pressurized because of coolant temperature rise due to postulated event as mentioned above, which in turn forces the molten metal kept in the tanks to flow into the gaps. When the flow starts, further flow of liquid metal takes place by syphon action. Figure 12 shows schematic of the gas gap filling system. The heat generated in the fuel gets conducted through the gap filled with liquid metal and gets dissipated from the body of the reactor block by natural convection to the surroundings.

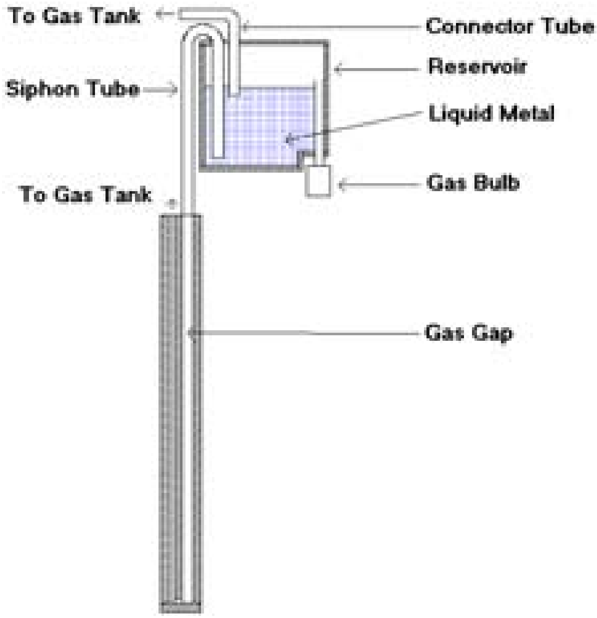

FIG. 12. Gas gap filling system of CHTR.

Figure 13 shows the steady state reactor block middle plane temperature distribution when the liquid metal core inlet and outlet temperatures are 1173 K and 1273 K respectively. The analysis has been carried out for 3-dimensional reactor geometry using finite element code ANSYS. In this case, mainly the main coolant flowing by natural circulation in the main heat transport loop removes the heat generated in the core. It can be seen from Fig. 13 that the major temperature drop is through the gas gaps. Hence, the gas gaps are highly effective and act as insulation.

The steady state analysis is further carried out to determine the amount of heat dissipated from the body during the loss of heat sink. It was assumed that in postulated accidental case mentioned above, the neutronically limited reactor power would increase and stabilize at 200 kWth. The entire heat loss is in radial direction by predominately conduction mode in the reactor block and natural convection in the water pool maintained at 323 K. Figure 14 shows the steady state temperature distribution in the reactor block due to the postulated event. The liquid metal may get solidified in the gas gap in long term after shut down. For restart of the reactor, this metal is melted and pumped back to the tanks by active means.

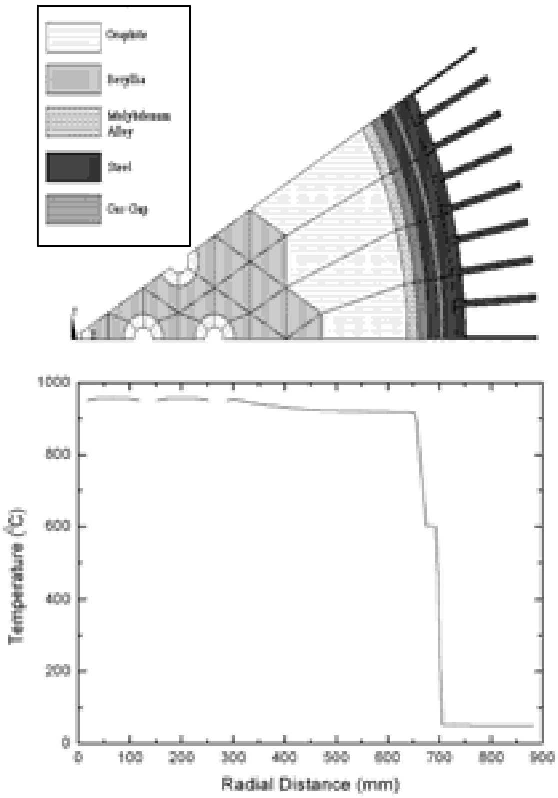

FIG. 13. Temperature distribution in reactor block in normal case.

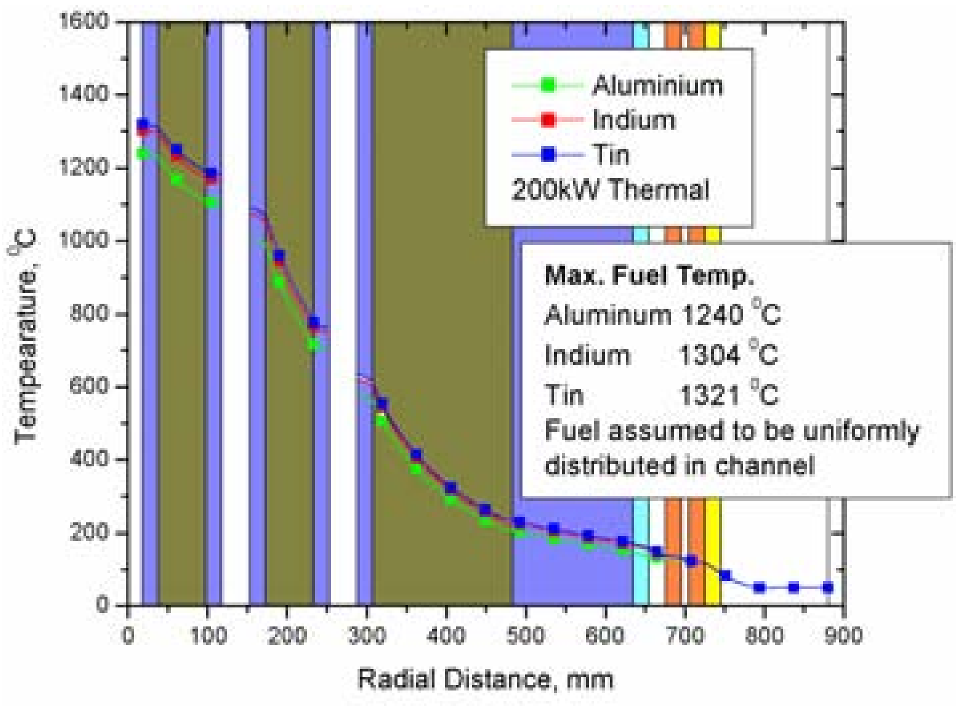

FIG. 14. Temperature distribution in reactor block during postulated accident.

3.2 Passive power regulation system

Passive regulation strategy is employed in the CHTR. It is based on the use of increase in core outlet temperature in addition to the use of reactivity feedbacks to provide inherent adjustment of the power level to match heat removal rate due to change in burn-up and in upset conditions. The passive power regulation system shown in Fig.15 has gas header at the top. Attached to the gas header is a niobium driver tube. The driver tube is housed in the control tube, which is surrounded by graphite sleeve. The graphite sleeve is an external fitment and does not interfere with the normal operation of the regulation system. The sole purpose of the graphite sleeve is to reduce heat transfer to the control tube. The gas header acts as the temperature sensor and is located in the top plenum, being submerged in the coolant. Under normal operating condition, the gas header is located in the region of coolant temperature approximately 900 K. Any condition, which causes the coolant temperature higher than the normal temperature will lead gas contained in the gas header to get heated up. Due to this, the temperature of the gas will rise leading to a rise in pressure in the driver tube. This causes a pressure imbalance between the driver and control tube and results in the absorber rod floating in the liquid to go further up into the core introducing negative reactivity. The behaviour of the passive power regulating system was analysed and shown capable of passively regulating the reactor power. For shut down of the reactor the gas header is externally pressurized.

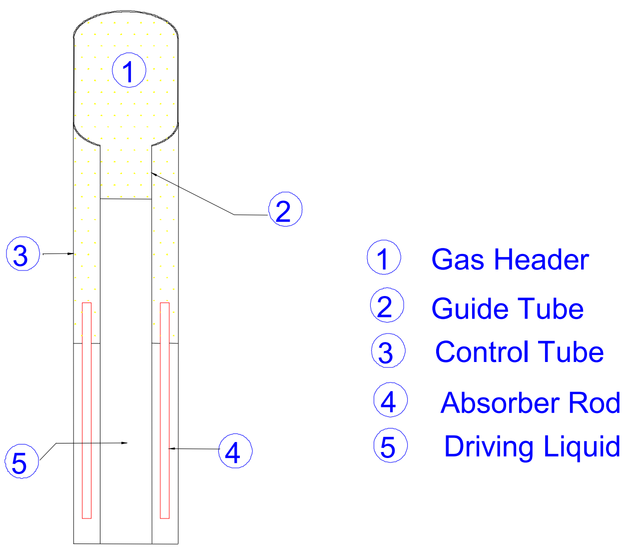

FIG. 15. Passive power regulation system.

CONCLUSION

Several inherent and passive systems have been adopted in Indian innovative AHWR and CHTR reactors. The analyses have been performed to prove design concepts of these systems. Experiments and further analyses of these systems are being carried out rigorously. Several major areas of R&D have been identified and the required development activities have been initiated.

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ANNEX 10

SMALL MODULAR LEAD-BISMUTH COOLED FAST REACTOR FOR MULTI-PURPOSE USE: SVBR-75/100

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Abstract. The paper presents an overview of the projects of reactor installations with small modular lead-bismuth cooled reactors SVBR-75/100. The R&D performed in the Russian Federation has demonstrated technical feasibility and potential economic competitiveness of the SVBR-75/100 reactor installations for nuclear power systems of both near and far future. In its present design, the reactor implements a conservative approach thoroughly based on 80 reactor-year experience in the operation of small lead-bismuth cooled reactors for nuclear submarines. Further on, upon construction and successful operation of an SVBR-75/100 demonstration prototype, more innovative approaches to reactor design could be gradually validated and implemented. Modular structure of nuclear steam supply system of a power unit with SVBR-75/100 reactor installations makes it possible to reduce the NPP construction period and, in the future, to make a transfer to standardized design of power units of different capacity on the basis of serially produced reactor modules. Such approach would assure competitiveness of the NPPs not only in electricity markets but in investment markets as well. It is shown efficient to use SVBR-75/100 reactor installations for renovation of the second unit of the Novovoronezhskaya NPP in the Russian Federation. It could also be envisaged that SVBR-75/100, a reactor that meets the requirements to nuclear power systems of the 21st century, will provide a basis for launching collaborative international project.

1. INTRODUCTION

Small lead-bismuth cooled fast reactors SVBR-75/100 discussed in this paper are based on actual experience in the development and operation of lead-bismuth cooled reactors for nuclear submarines [1]. In fifteen-twenty years from now it will be possible to deploy SVBR-75/100 in both industrialized and developing countries. These reactors make it possible to resolve a contradiction between economic characteristics and safety requirements that is peculiar to reactors of traditional type. Due to their improved technical and economical characteristics and higher safety level, fast reactors with lead-bismuth coolant could be considered as one of the possible candidates for step-by-step replacement of thermal reactors [2].

The technical basis for SVBR-75/100 reactor installation design is as follows:

- 50-year experience in the design and operation of reactor installations with heavy liquid metal coolant for nuclear submarines;
- experience in the construction and operation of sodium cooled fast reactors;
- experience in the validation of heavy liquid metal coolant technology in the reactors of nuclear submarines and ground-based facilities;

- design drawings of SVBR-75/100 nuclear installations for the renovation of units 2, 3, and 4 of the Novovoronezhskaya NPP (NVNPP), conceptual design of an NPP with two 1600 MW_e units based on reactor installations of SVBR-75/100 type.

2. SVBR-75/100 CONCEPT

The design features of SVBR-75/100 are targeted at the use of the industrial base, structural materials, and verified technology of lead bismuth coolant as available in Russia.

Conservative approach was used in the design of SVBR-75/100 (lead-bismuth fast reactor of 75-100 MW equivalent electric power, depending on steam parameters), which provides for the retention of operation parameters of the primary and secondary circuits at the levels that have been already mastered in practice, for the use of qualified fuel and structural materials and proven technical solutions for the equipment components and reactor installation scheme. This approach assures that technological approaches implemented in other reactor installations, first of all, the propulsion reactors of nuclear submarines, have been to the maximum possible extent inherited by the SVBR-75/100 design. Adhering to this approach reduces the terms, scope and costs of necessary R&D and investment risk, and secures high reliability and safety of the reactor installation.

The application of conservative approach does not mean that new technical solutions could not be implemented, and that only an evolutionary approach should be applied to NPP design. Be it so, this would cause stagnation of scientific and technical progress. However, as the use of practically verified technical solutions ensures the applicable technical and economical parameters of SVBR-75/100 NPP [2], further improvements can be realized in a step-by-step mode when changing over to the next generation of installations of a given type.

In line with the above discussion, SVBR-75/100 design incorporates the following basic approaches and technical features:

- pool type integral (mono-block) design of primary circuit equipment with complete elimination of the pipelines for lead bismuth coolant and valves;
- two-circuit scheme of heat removal;
- normal operation functions and safety functions in the reactor installation systems are combined to the maximum possible extent;
- levels of natural circulation in heat-removal circuits are sufficient to secure reactor after-cooling;
- the reactor mono-block with a safeguard vessel is installed and fixed in the tank of the passive heat removal system (PHRS). The tank is filled with water and also shoulders the function of neutron shielding;
- upon the end of fuel lifetime, the refuelling can be performed at once, assembly by assembly;
- different types of fuel, e.g. UO_2, mixed oxide (MOX) fuel with weapon-grade or reactor Pu, MOX fuel with minor actinides, nitride fuel can be used without changing the reactor design and configuration and without violating the requirements to safety;
- modular design of the mono-block's basic elements, option to replace and repair these elements separately;
- small weight and size the mono-block secure an option of its factory fabrication and transportation to the NPP site by any mode.

With due account of the higher cost of lead bismuth coolant as compared to that of other liquid metal coolants, measures to reduce the specific mass of lead bismuth have been developed and implemented in the design. Analysis of the experience in the development of power reactors of various types [3] has revealed that specific mass of the coolant decreases with the reduction of reactor's nominal power. On the other hand, there are factors limiting the reduction of lead bismuth coolant mass. Thus, when core dimensions are small, it is impossible to secure that core breeding ratio (CBR) is equal to 1. Calculations have shown that core diameter should be not less than $1600 \div 1700$ mm at 900 mm height. Such dimensions make it possible to achieve ~ 100 MW equivalent electric power of the reactor. In this case, CBR $\cong$ 1 is achieved not only for the mixed nitride fuel but also for the less dense but well mastered MOX fuel. Such CBR can only be assured if the volumetric fraction of fuel is not less than $55 \div 60\%$.

Low specific mass of lead bismuth coolant in small power fast reactors with the specific core power density several times lower than that in sodium cooled reactors is achieved through the elimination of in-vessel repository of spent nuclear fuel and in-vessel refuelling mechanisms (rotating plugs, etc).

Another way of reducing the specific mass of lead bismuth coolant is to increase its average flow rate and to diminish the length of circulation circuit. However, this approach has its own constraints caused by the necessity to meet safety requirements. The first requirement is defined by the necessity to provide the power level of the reactor with naturally circulating lead bismuth coolant at the level not less than 5…7% of its nominal power. This makes it possible to eliminate inadmissible temperature increase under a shutdown of main circulation pumps. The second requirement is conditioned by the necessity to secure conditions for the assured surfacing of steam bubbles from lead bismuth coolant to its free surface level under the rupture of steam generator (SG) tubes. This is important to eliminate steam ingress into the core and inadmissible pressure increase in the mono-block vessel.

The necessity to meet the highlighted requirements has resulted in the development of a circulation scheme, in which the core hydraulic resistance equals to 90% of the total hydraulic resistance of primary circuit, while the hydraulic resistance of the SGs, in which lead bismuth coolant flow rate is much smaller, equals only to 10% of the total value. With due account of the listed requirements, the specific mass of lead bismuth coolant in SVBR-75/100 reactor installation is ~ 1100 t/GWe.

The selected power level makes it possible to secure:

- a lifetime of ~ 53000 effective hours when well-mastered uranium dioxide fuel is used (CBR = 0.87);
- CBR ≥ 1 when MOX fuel is used; in this case the reactor can operate in a fuel self-sufficient mode within a closed fuel cycle;
- CBR ≥ 1 when mixed nitride fuel is used; in this case the reactor can operate in a fuel self-sufficient mode within a closed fuel cycle and assure the burn-up reactivity swing to be less than β_{eff}, or it can operate in a breeder mode with CBR = 1.13 and the plutonium doubling time of ~ 45 years;
- the burn-up reactivity swing less than β_{eff}, and the lifetime duration of ~ 80000 effective hours when uranium nitride fuel is used;
- complete factory fabrication of the reactor mono-block; reactor installations can be mass-produced serially, which would improve their quality and reduce their cost;
- an option to transport the reactor mono-block by railway, by truck, or by marine

transport with its fuel being in a nuclear- and radiation-safe state due to the “freezing” of lead bismuth coolant in the mono-block vessel (this may also enhance proliferation resistance);

- an option to renovate ageing NPP units by replacing the original reactor installations with new ones once in 50-60 years. Then, the construction of the replacing power capacities may be postponed by ~50 years;
- the decommissioning cost of the unit considerably reduced, since there are no radioactive materials in the main reactor building after the reactor mono-block is removed.

3. DESIGN DESCRIPTION OF SVBR-75/100

The principal scheme of SVBR-75/100 is presented in Fig. 1, showing all basic systems of the reactor installation.

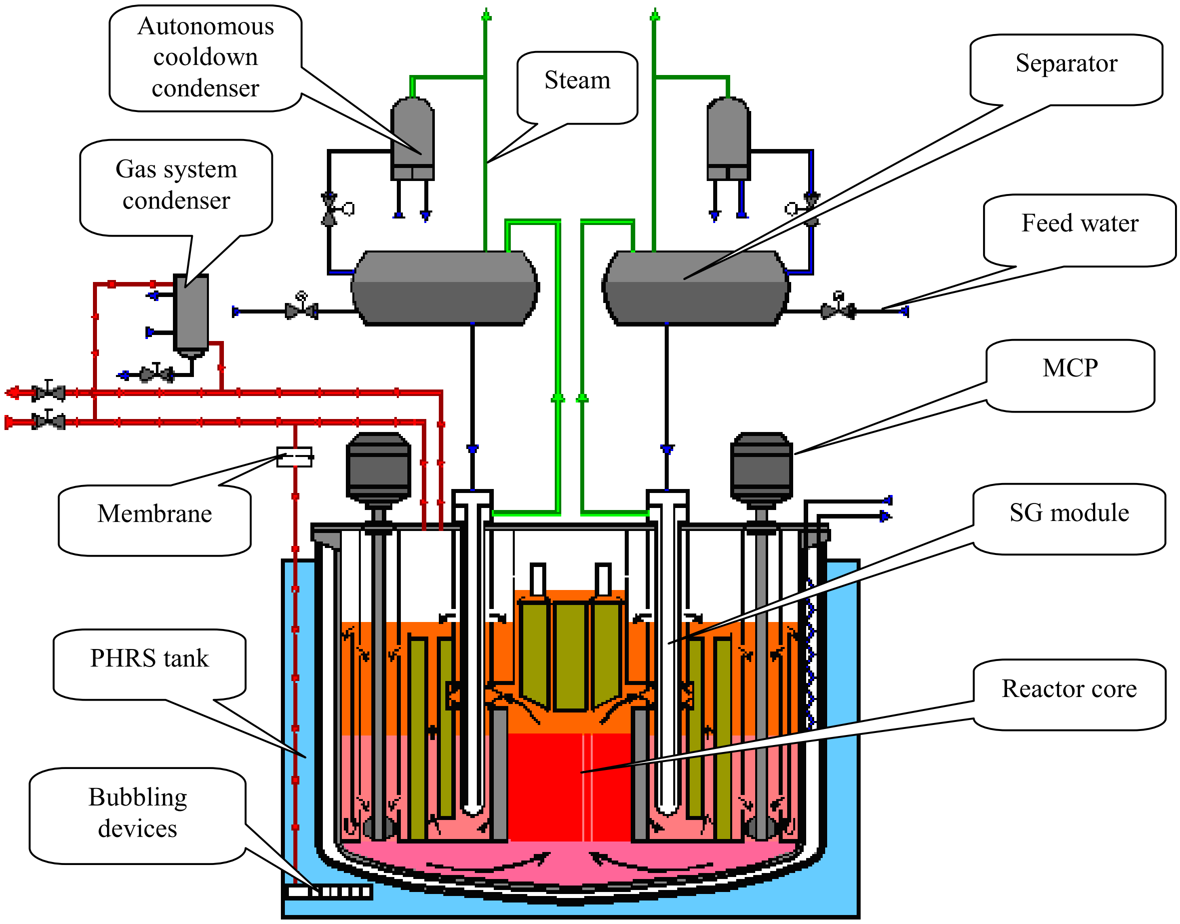

FIG. 1. The principal hydraulic scheme of SVBR-75/100.

The primary circuit includes core, steam-generator (SG) modules, main circulation pumps (MCPs) and in-vessel radiation shielding, all installed in a reactor mono-block vessel.

The secondary circuit includes SG modules, feedwater and steam pipelines, separators and autonomous cooling condensers.

The gas protection system includes condensers, protection membrane device, bubble device and pipelines.

The heating system is designed for heating of the reactor mono-block prior to filling it with coolant and for keeping it in a hot state. It includes a system of pipelines installed between the basic and guard vessels of the reactor mono-block. These pipelines supply steam for heating.

The coolant technology system includes mass-exchangers, ejectors of gas mixture, and sensors of oxygen activity in lead bismuth coolant. This system is designed to maintain certain lead bismuth quality, which is necessary to prevent the corrosion of structural materials.

Safety systems are represented by emergency protection system of the reactor (EP), system of leak localization in the SGs, autonomous cooling system (ACS), and passive heat removal system (PHRS). Of them only EP is a dedicated safety system, while other systems, namely the system of leak localization in the SGs, ACS, and PHRS combine the functions of normal operating systems and accident prevention systems.

The basic parameters of SVBR-75/100 are given in Table 1.

TABLE 1. BASIC CHARACTERISTICS OF SVBR-75/100

Parameter	Value
Thermal power, MW	280*
Electric power, MW	101.5*
Steam production rate, t/h	580*
Steam parameters: pressure, MPa temperature, °C	9.5* 307*
Feedwater temperature, °C	241*
Lead bismuth temperature in primary circuit, °C: core inlet core outlet	482* 320*
Core dimensions (diameter × height), m	1.645 × 0.9
Average volumetric power density of the core, kW/dm^3	140*
Average linear heat rate of fuel element, kW/m	~ 24.3*
Fuel (UO_2): U-235 load, kg U-235 enrichment, %	~ 1470* 16.1*
Core lifetime, thousand effective hours	~ 53
Interval between refuelling, years	~ 8
Number of SGs	2
Number of SG modules	2 × 6
Number of MCPs	2
Power of MCP electric driver, kW	450
MCP head, MPa	~0.55
Lead bismuth coolant volume in primary circuit, m^3	18
Reactor vessel dimensions (diameter × height), m	4.53 × 6.92

*The characteristics correspond to SVBR-75/100 configuration for a modular NPP with two units of 1600 MW_e each. These characteristics may be changed if SVBR-75/100 is to be used as a component of other NPPs.

In addition to this, the reactor installation includes fuel handling system that is a set of equipment (an adapter box with gate valve, refuelling pressure suits for extracting the shielding plug and the core basket, and refuelling container) for the reloading of spent fuel sub-assemblies into capsules filled with lead and for the installation of a new core basket with fresh fuel.

The basic equipment of SVBR-75/100 is installed in a tight-box confinement of 11.5 m height (Fig.2).

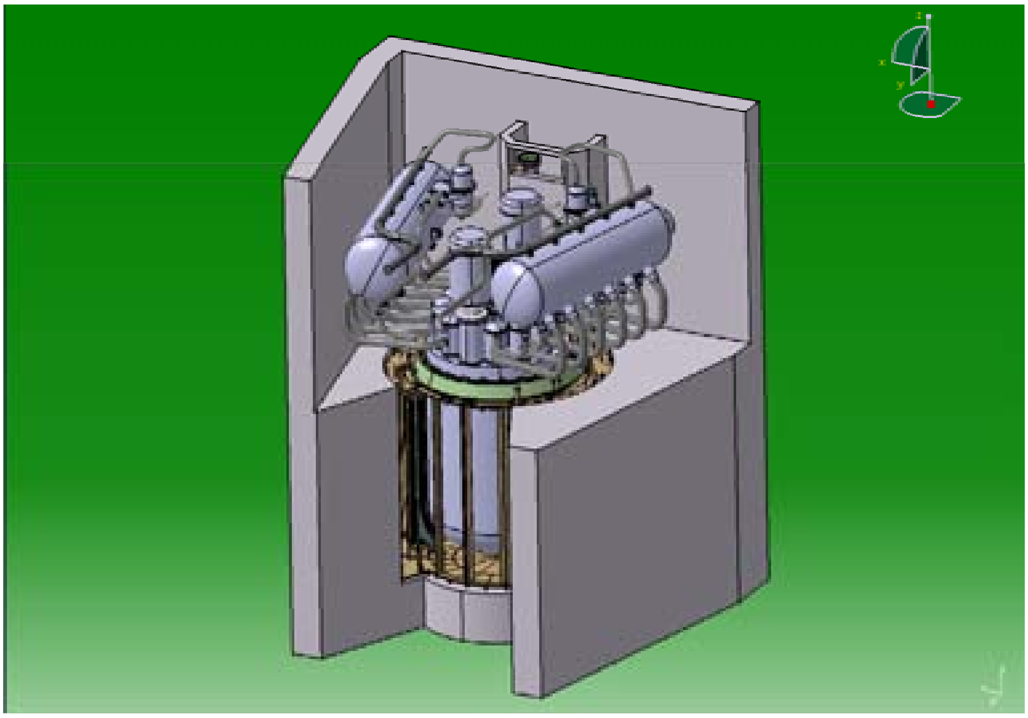

FIG. 2. Arrangement of SVBR-75/100 equipment.

In its lower part, each box has a concrete well for the PHRS tank to be mounted. The reactor mono-block is installed inside the PHRS tank and is fastened to the head ring of the tank lid. PHRS tank also accommodates 12 immersible heat exchanges, which transfer heat from the PHRS tank to the intermediate circuit water.

In the upper part of the box, above the PHRS tank, mounted are the components, which do not belong to the design equipment of the reactor mono-block. These are two steam separators and two cooling condensers. The height mark of separators' location has been selected to provide the necessary level of natural circulation in the secondary circuit in all modes of reactor operation.

Condensers of the gas system are installed in the upper part of the box in, a separate concrete compartment.

The hatches designed to withstand extreme loads are mounted over each reactor installation to facilitate assembly, repair, and maintenance operations as well as refuelling.

All equipment components of the primary circuit are installed inside a strong vessel of the reactor mono-block. In its central part, the vessel hosts a removable unit that includes a basket with the core, control rods, and a shielding plug and is surrounded by the in-vessel radiation shielding with the SG and MCP modules mounted on it (see Fig. 3).

The paths for coolant circulation in the two circuits (the main and the auxiliary one) of the primary system are formed exclusively by the components of the in-vessel devices within the reactor mono-block vessel without any pipelines and valves being used.

Within the main circulation circuit, the coolant flow is organized according to the following scheme. Being heated in the core, the coolant flows to the inlet in the medium part of the inter-tube chamber of the twelve SG modules switched on in parallel. Then, the coolant is divided into two flows. The first flow moves bottom-up in the inter-tube chamber and reaches the peripheral buffer chamber, which has a free level of the "cold" coolant. The second flow moves top-down and reaches the outlet chamber from which it is directed to the channels located within the in-vessel radiation shielding. It is cooled when moving up and then is also directed to the peripheral buffer chamber. From the peripheral buffer chamber the main coolant flow goes over the downcomer circular channel along the reactor mono-block vessel and through the inlet chamber to the MCP suction. Another part of the coolant flow gets to the MCP suction over the circular channel formed by the vessel and the MCP shaft. Out of the MCP the coolant is transferred over the two channels organized in the block of the lower zone of the in-vessel radiation shielding into the distributing chamber, from which it is directed to the reactor inlet chamber, thus closing the main circulation circuit.

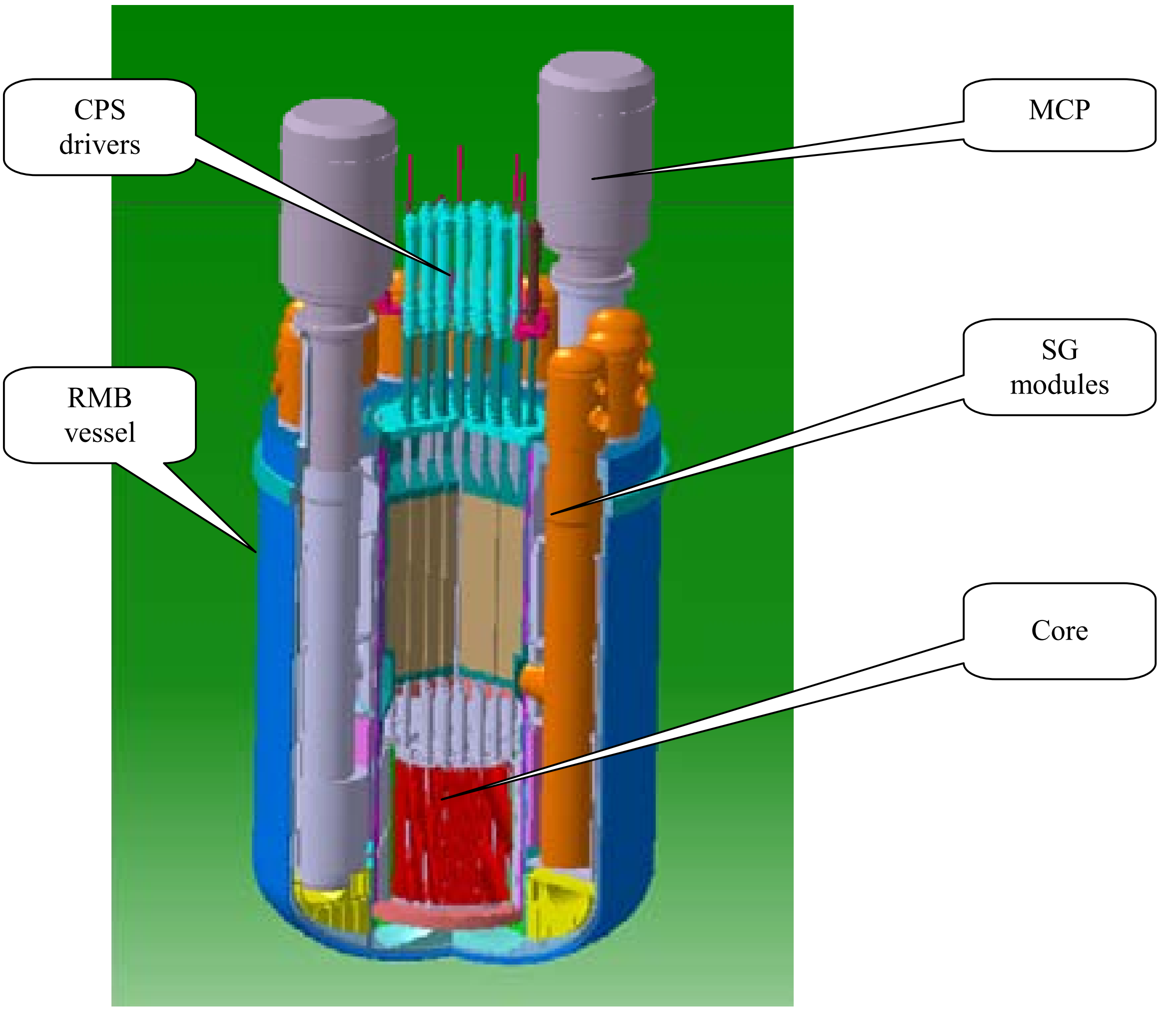

FIG. 3. Arrangement of the equipment in reactor mono-block vessel.

The circulation scheme adopted for the main circulation circuit with free levels of coolant in upper part of the reactor mono-block and SG module channels together with low flow rate of the coolant in downcomer sections secures the reliable separation of water admixtures from coolant in the event of an accidental tightness failure of the SG pipes.

The auxiliary circulation circuit is organized through the channels originally designed for mounting of the jackets of the control and protection system (CPS) and through the channels in the reactor's shielding plug. It shoulders the functions of cooling of the CPS absorbing rods and provides the required temperature levels in central buffer chamber and in mass exchanger channels.

4. CONCEPT OF A MODULAR NPP BASED ON SVBR-75/100 REACTOR INSTALLATIONS

Reduction of the investment cycle for NPP construction through application of a modular approach with the delivery of factory-fabricated and ready modules may be important to make technical and economic parameters of an NPP comparable to those of a steam-gas cycle heat power plant (HPP). For developed countries, with their power systems being based on high-voltage transmission lines, the use of large power modular units may be economically effective. Maximum possible capacity of a modular type power plant should not be limited by maximum possible capacity of the reactor.

The Russian Federal State Unitary Enterprises: the State Scientific Centre "Institute of Physics and Power Engineering" (IPPE), the Experimental Design Bureau (EDO) "Gidropress", and "Atomenergoproekt" have developed conceptual design of a two-unit NPP with each of its units being based on a nuclear steam-supply system (NSSS) consisting of 16 SVBR-75/100 reactor modules and a single turbine installation of 1600 MWe [2].

When selecting the NPP unit capacity, it was taken into account that specific capital costs of the reactor compartment would decrease at increasing the unit's capacity. It is due to the fact that at increasing the number of modules in the reactor compartment, the cost of the equipment and providing systems installed beyond it increases only slightly. For this reason, their contribution to the specific capital costs of the reactor compartment will decrease. Such systems and equipment include the refuelling equipment, the equipment for coolant intake, the equipment for transfer of the coolant to reactor mono-blocks at initial filling, radioactive waste handling system, etc.

Modular principle of NSSS design is economically more effective for the reactors, in which inherent safety features to prevent severe accidents have been realized to the maximum possible extent. First of all, this is true for LOCA. To cope with such accidents, light water reactors (LWRs) need a lot of safety systems that are not necessary for SVBR-75/100. Elimination of such systems significantly simplifies the technology of assembly and construction and reduces the volume of construction for the reactor compartment.

An operator who uses a common power master unit carries out NSSS control. If there is any fault in a certain reactor installation, it is automatically removed from the operation and can be cooled down autonomously with the use of the turbine installation systems.

Modular design of NSSS together with infrequent refuelling makes it possible to achieve load factor not less than 90%. When reactor installations are one after another shut down for refuelling, this reduces the unit power only slightly.

Licensing of a large power modular unit construction will be much more simplified once SVBR-75/100 equipment is certified and SVBR-75/100 prototype is constructed. Small power of the reactor installation defines a comparatively low cost of its construction. The results of the technical and economic analysis have shown that, in compliance with the data obtained at the conceptual design stage, the technical and economical parameters of an NPP with two 1600 MWe units, each based on the SVBR-75/100 type reactor installations, are better than those of an NPP based on large capacity LWRs and than those of a steam-gas HPP with 10 PGU-325 units operating on natural gas.

The comparable characteristics of power plants of different types are summarized in Table 2.

TABLE 2. CHARACTERISTICS OF POWER PLANTS OF DIFFERENT TYPES

Characteristic	NPP with SVBR-75/100	NPP with VVER-1500	NPP with VVER-1000	HPP with PGU-325
1. Installed capacity of a unit, MWe	1625	1479	1068	325
2. Number of units	2	2	2	10
3. Electric power for plant's own needs, %	4.5	5.7	6.43	4.5
4. Net efficiency, %	34.6	33.3	33.3	44.4
5. Specific capital investments in plant construction, $/kWe (prices of 1991)	661.5*) 563**)	749.8	819.3	600
6. Projected electricity cost, cent/kWh (prices of 1991)	1.46	1.85	2.02	1.75

*) With the additional margin of ~17% over the normative cost being introduced, which is 60% of the cost of SVBR-75/100 equipment.

**) With due account of an option to make a transfer to the superheated steam in turbine circuit and to increase the temperature of fuel element claddings up to 650˚C.

5. FUEL CYCLE AND SPENT NUCLEAR FUEL MANAGEMENT

Due to the low current costs of natural uranium and enrichment, the use of uranium oxide fuel with the spent nuclear fuel (SNF) storage at NPP site and postponed reprocessing is economically justified for SVBR-75/100. The duration of such storage depends on the available uranium resources and the scale of nuclear power. Anyway, the existing uranium resources are sufficient to realize a realistic scenario of nuclear power development till the year 2050. It can be expected that the costs of natural gas at the domestic market will increase more intensively than the costs of natural uranium. This will ensure the competitiveness of NPPs even in the case of a considerable increase of uranium prices, because the structure of electricity costs for NPPs and HPPs is essentially different.

As experience in the operability of core elements is already available, the major way of improving the economic parameters of SVBR-75/100 fuel cycle at this stage will be extending the core lifetime (increasing fuel burn-up). Further on, at the initial stage of own SNF reprocessing, only uranium will be recycled, while plutonium, minor actinides and fission products will be extracted and then stored, as would be economically expedient. Duration of the uranium stage may be extended when changing over to the uranium nitride fuel.

Actually, in the future it will be necessary to change over to the entirely closed nuclear fuel cycle. The time period necessary for this change will be determined by economically effective SNF reprocessing that has to be developed at an industrial scale and should be acceptable from the standpoint of radioactive waste minimization and non-proliferation. Change over to the closed NFC will allow economically effective use of the SNF from thermal nuclear reactors (TRs) as a make-up fuel for SVBR-75/100. As the fraction of the TRs' SNF in the fresh fuel of SVBR-75/100 operating in a closed fuel cycle is ~ 10-12%, and the plutonium fraction in the TRs' SNF does not exceed ~1%, the effect of the plutonium isotopic vector in the SNF of both VVER and RBMK reactors on the isotopic vector of the fresh fuel for SVBR-75/100 will be negligible. Therefore, in the future SVBR-75/100 will make it possible to develop a principally new strategy of the closed nuclear fuel cycle that would not require expensive reprocessing of the TRs' SNF for the extraction of only ~1% of plutonium for its further use in fast reactors.

The flexibility of SVBR-75/100 in relation to nuclear fuel cycle that may be realized in line with the principle: "to operate using the type of fuel that is most efficient at the moment" makes it possible to postpone the task of constructing a specialized fuel cycle factory by several decades after the first unit of an NPP with such reactors is launched. For example, after the introduction of about 10 GWe of power capacities based on SVBR-75/100 and after getting the NPP construction costs repaid, a certain share of the profit could be spent on launching the industry for SNF reprocessing and fabrication of the fuel sub-assemblies from mixed fuel. Such factory being launched, only the current operating costs of SNF reprocessing and fuel sub-assembly fabrication would define the cost of SVBR-75/100 core. If the developments of the Russian State Scientific Centre "Research Institute for Atomic Reactors" (RIAR) on pyro-electric-chemical SNF reprocessing in chloride melts [5] are used as a basis of that complex, the contribution of fuel costs to the cost of SVBR-75/100 core would be even less than that of the basic variant using the uranium oxide fuel. This would make it possible to improve the NPP competitiveness considerably. Such approach to the construction of capacities for SNF reprocessing and fuel sub-assembly fabrication presumes that the owner of the NPP units is also the owner of fuel cycle factory.

The following procedure is provided for SNF storage before reprocessing. After spent fuel sub-assemblies are extracted from the reactor, they are installed in the capsules in which lead has been preliminary heated in the electric oven up to the temperature exceeding its melting point. Then the capsule is sealed and transported to a "dry" repository with natural air-cooling. Very soon lead in the capsule solidifies and, therefore, four barriers are formed on the way of radioactivity release to the environment: fuel matrix, fuel element cladding, solidified lead, and capsule shell. The solidified lead that contacts steel cladding of a fuel element eliminates the corrosion effect.

When reprocessing SNF, it is assumed that the extracted fission products first are vitrified and then, after necessary cooling, are enclosed in special containers providing a multi-barrier shielding and transported to be finally disposed in deep geological formations. Minor actinides (except for curium) are not separated from plutonium and are used in the reactor as a fuel component. Curium is extracted and transported to the temporary repository for 100-150-year cooling. Upon being cooled, all curium isotopes (except for curium-245) are transformed into plutonium isotopes. Then this isotopic mixture is used to produce new fuel for the reactor.

6. INTRINSIC PROLIFERATION RESISTANCE BARRIERS

Intrinsic proliferation resistance barriers assume creation of the conditions when inappropriate use of fissile materials is the least attractive for the nuclear weapon's potential distributors. It is evident that the problem of non-proliferation cannot be solved only by technological measures, as there are opportunities for illegal production of the weapon materials by using the well-developed technologies of uranium isotopic separation and plutonium extraction from spent nuclear fuel. Complete resolution of non-proliferation problem may be achieved only through the combination of technological features and verification and political measures. During the recent decades all nuclear countries that legally possessed nuclear weapons were successful in solving this problem by the implementation of physical protection measures, accounting, control, and safeguards. For this reason, additional proliferation-resistance features achieved through specific technological approaches will be justified only if they do not diminish the competitiveness of nuclear power. For developing countries, intrinsic proliferation resistance features of a certain technology should be implemented along with relevant political measures and measures of international control.

Fuel transportation within the reactor mono-block with solidified lead bismuth coolant creates an additional technical barrier to prevent fuel theft. Solidified lead bismuth coolant in the reactor mono-block also eliminates the risks of nuclear and radiation accidents in transportation. It may be expedient to concentrate SNF reprocessing at certain factories. In this, technological support of the non-proliferation regime may be provided through the application of the process of SNF reprocessing, in which 2% of fission products and all minor actinides remain in the re-fabricated fuel. The accounting and control of such fuel is simplified, because its handling requires special equipment.

When uranium fuel is used, using uranium fuel with the enrichment less than 20% ensures compliance with the non-proliferation regime and by securing that SNF is stored under the protection of fission products' gamma-radiation ("Spent Fuel Standard").

7. CONCLUSION

The R&D performed has demonstrated technical feasibility and potential economic competitiveness of the SVBR-75/100 reactor installations for nuclear power systems of both near and far future. The modular structure of NSSS of a power unit with SVBR-75/100 reactor installations makes it possible to reduce the NPP construction period and, in the future, to make a transfer to the standardized design of power units of different capacity on the basis of the serially produced standard modules offering a broad spectrum of inherent safety features. Such approach will assure competitiveness of the NPPs not only in electricity markets but in investment markets as well. Power units with SVBR-75/100 could be used in both developed and developing countries. For SVBR-75/100 it is possible to use different types of fuel and to operate reactor in different fuel cycles, preferably the ones that turn to be more efficient at certain moments of nuclear power evolution. When operating under a closed nuclear fuel cycle, it is possible to assure fuel self-supply regime or to provide a small breeding. The SNF of thermal nuclear reactors may be utilized as a make-up fuel for SVBR-75/100.

The highlighted options could be facilitated by the construction of SVBR-75/100 prototype plant. It will be efficient to use SVBR-75/100 reactor installations for renovation of the second unit of the Novovoronezhskaya NPP in the Russian Federation.

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THE STAR CONCEPT: A HIERARCHICAL HUB-SPOKE NUCLEAR ARCHITECTURE BASED ON LONG REFUELING INTERVAL BATTERY[1] REACTORS AND REGIONAL FUEL CYCLE CENTERS

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Abstract[2]. The STAR reactor and fuel cycle concept is devised to attain Gen-IV goals by responding to foreseen mid century needs and market conditions. It is targeted to fill energy and potable water needs for urban centres in developing countries and is designed to fit within a hierarchical hub-spoke energy architecture based on regional fuel cycle centres, using nuclear fuel as the long distance energy carrier – with distributed electricity generation as the local carrier to mesh with existing urban energy distribution infrastructures using grid delivery of electricity, potable water, and communications (and sewage return) through a common grid of easements. STAR is also intended for Independent Power Producers in industrialized countries seeking to service emerging markets for hydrogen and water production. STAR concept development is being conducted for a portfolio of specific reactor and balance of plant designs to enable an incremental market penetration that is time-phased according to the degree of R&D required. STAR-LM is a Pb-cooled, 400 MW_{th}, natural circulation reactor of 565°C core outlet temperature driving a supercritical CO_2 Brayton cycle for electricity production. It draws on many proven technologies and will be ready for market in 15-20 years. STAR-H_2 raises the Pb outlet temperature to 800°C to drive a thermochemical water cracking cycle and will require additional R&D. SSTAR takes the STAR-LM design features down to ~25 to 50 MW_{th} for secure energy supply to remote small villages. All STAR concepts are designed for 20-year refuelling interval and rely on outsourcing fuel cycle and waste management services to proposed regional fuel cycle centres. All employ desalination (or alterative) bottoming cycles to extend their scope of energy services and to minimize their environmental footprint.

1. BACKGROUND & GOALS

In his prescient plenary speech "On the nature of nuclear power and its future" [1] at the GLOBAL 93 conference, Wolf Häfele compared the technical, institutional, and social opportunities for a second wave of nuclear deployments to those which brought about the Industrial revolution. He argued that the first wave of nuclear deployments – for electricity production and with an open fuel cycle – was *destined* to saturate at under 400 GWe global deployment because:

> "Nuclear power was put into an existing technical and institutional infrastructure without much changing this infrastructure – still characterized by the use of oil in particular but also of coal and gas" [i.e., nuclear was deployed in an energy supply architecture optimised for fossil].

But to paraphrase his view of the analogy:

> The Industrial revolution exploited the factor of a million between ~1 μ eV due to mass flow (of falling water) and ~1ev chemical energy flow of burning coal to achieve a revolutionary transition away from centuries of reliance on water wheel and animal power to coal-fired steam engines. The exploitation of the factor of a million between renewable and chemical energy density achieved by changing over to a *stored* (coal) resource – *when enabled by re-engineering the architecture of production (factories,*

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[1] STAR = Secure, Transportable, Autonomous Reactor. The STAR reactors are referred to as "Batteries" because they store 20 years worth of heat and they load follow by passive means – delivering heat when it is requested by the Balance of Plant and passively shutting off when the request stops.

[2] This paper has been assembled from several previously published papers – often with verbatim copying of textual material and figures. The prior papers are cited in the list of references – especially 6, 7, 8, 12, and 16.

> *division of labour, etc.)* – led to the first Industrial revolution. Over the ensuing one and a half centuries this revolution literally changed the Western world [technically, institutionally, and socially] and finally broke the Malthusian regression to unchanging GDP/capita.

He concluded:

> "One must be prepared for evolution or even revolution when *real* nuclear power [frees itself from the architecture optimised for fossil, and] brings the factor of a million between nuclear and chemical bond energies to the surface – *one cannot treat nuclear power like chemical power, -- uranium like yellow coal, so let us not loose our perspective*."

Nuclear energy's current configuration has left its most important innate features unexploited; its economically harvestable resource base good for a millennium of global energy supply by closing the fuel cycle; its capacity to service the *entire* primary energy market by manufacturing hydrogen; its capacity to break the energy security/non-proliferation dilemma by exploiting its incredible energy density – serving as a long distance energy carrier to fuel long refuelling interval reactors supported by regional fuel cycle centres operating under international oversight, and its ability to achieve neutral radiological exchange with the ecosphere in the long term by self consuming its very long lived radioactive waste.

The objective of the work reported here has been to exploit these features to propose a re-engineered world energy supply architecture optimised for nuclear rather than fossil. It is intended for global energy supply in the market conditions of mid 21st century and beyond where 80 percent of the world's population of ~10 billion people reside in cities; where energy use per capita will have increased worldwide; where electricity and hydrogen serve as complementary energy carriers replacing fossil; and where ecologically-neutral closure of the world's energy supply chain is attained by eliminating carbon from the chain and by sending only fission products to waste. This nuclear-based architecture is intended to fuel a transition to global sustainable development.

Although it is revolutionary in concept, it is aimed to gradually displace the fossil architecture and manage the back end of the current nuclear infrastructure over a four or five decade evolutionary market penetration process.

A portfolio of STAR reactor and power plant concepts to fit within the proposed architecture are under development by a team of US national laboratories (Argonne, Livermore, and Los Alamos National Laboratories). The portfolio (described in subsequent sections) facilitates time phasing of market entry of a sequence of STAR concepts, each based on more aggressive R&D outcomes than the previous one. All rely on the same concept of regional fuel cycle centres and a hierarchical hub-spoke architecture of energy carriers organized in the order of their energy density and power transmission through practical sized energy transfer conduits. In this paper the architecture is described first; then the power plants are described.

Just as was the case for the Industrial revolution, technology by itself is not sufficient; the transition to nuclear-fuelled sustainable development in the 21st century will require institutional as well as technological innovations. These institutional innovations are discussed at the end.

2. MID CENTURY ENERGY NEEDS

Global energy demand forecasts for the 21st century project massive growth in demand for energy services, and they show that the dominant capacity additions by 2030 and beyond will occur in the currently developing economies. They predict that demographic migrations will lead to a majority of global population living and working in urban centres by mid century. Thus, the global reach of the nuclear client base must be expanded to include cities in developing countries. The range of demanded energy products will also expand; an emerging need for process heat conversion of water or hydrocarbon feedstocks to hydrogen is foreseen. Manufacture of potable water may also be needed as cities increasingly outsource municipal water supply contracts to profit-making entities.

Developing economies enjoy the opportunity to "leap frog" to new sustainable energy infrastructures, which meet their special needs. Population and economic activity, which is focused primarily in cities, will require an energy supply architecture having high energy density. Rapid economic growth rates will require emplacement of energy infrastructures having a short energy payback period. These two requirements preclude a major role for renewables for the developing country urban clients targeted here, but are well suited to the innate features of nuclear.

3. RECONFIGURING THE WORLD'S ENERGY ARCHITECTURE TO EXPLOIT NUCLEAR'S INNATE FEATURES

Nuclear has much to offer to fuel a sustainable development [2] revolution on the scale of the Industrial revolution, but to do so it must be reconfigured to meet the 21st century market situation. The fact that much of future growth will be in cities of developing nations means that market conditions facing future nuclear deployment will be different from historical conditions where deployment occurred primarily in industrialized countries under regulated electricity market conditions. The proposed STAR energy supply architecture has been optimised to exploit all of nuclear energy's innate features for the new market situation. STAR-LM and SSTAR power plants will generate electricity and potable water during the first several decades of market penetration. STAR- H_2 is intended for somewhat later when hydrogen joins electricity as a carbon-free energy carrier; STAR- H_2 is a 400 MW_{th}, turnkey plant that manufactures hydrogen, electricity, and potable water. STAR plants are targeted for worldwide deployment and especially for urban centres in developing countries – using nuclear fuel and hydrogen as the long distance energy carriers – and supporting distributed electricity generation as the local energy carrier. In that way the new architecture will *mesh seamlessly with existing and imminent urban energy distribution infrastructures using grid delivery of electricity, hydrogen, potable water, and communications (and sewage return) through a common grid of easements.* This will facilitate incremental market penetration. The small sizing and outsourced fuel cycle and waste management configuration allows for plant deployment at modest initial capital outlay for the client. Turnkey plants are transported to the client's site and rapidly connected to a pre-constructed non-nuclear safety grade balance of plant to achieve a rapid start of the revenue stream.

To break the energy security/non-proliferation dilemma, STAR plants are designed with 20-year refuelling interval and they fit within a proposed hierarchical hub-spoke energy supply architecture using regional fuel cycle centres. The regional centres handle both front and back end fuel cycle services – including waste management. They are under the operational control of consortia of regional clients and operate under international non-proliferation oversight. Regional centre personnel using relocatable refuelling equipment, which they bring to the STAR site to conduct refuelling operations and then remove and take away with the used cassette, conduct whole core cassette refuelling operations on a 20-year refuelling interval.

Figure 1 illustrates the proposed hierarchical energy delivery infrastructure at an abstract level. The “hubs” represent where one energy carrier (nuclear fuel, hydrogen, electricity) is converted into the successive energy carrier along the supply chain – a carrier better suited to the required function. The “spokes” represent the transmission channels of the energy carrier from its source point to its point of use. The ordered sequence of energy carriers (nuclear fuel shipped from the regional centres to the battery nuclear power plants sited near a city’s perimeter; hydrogen and water piped from the STAR nuclear plants to the district load centres; and electricity wired from distributed production centre to end use) *are organized sequentially (hierarchically) in the order of their energy density and their associated power carrying capacity through practical-sized conduits (e.g., ships/trains; pipelines/trucks; wires, respectively).* The widths of the spokes in Fig. 1 suggest the power carrying capacity of practical conduits for each energy carrier; the fractal-type expansion of the architecture as it progresses from the uranium ore energy resource to the point of end energy use reflects the diminishing energy carrying capacity and corresponding multiplicity of carrier conduits in the hierarchical sequence of energy carriers.

For example, e.g., a two-week voyage to deliver a single 400 MW_{th} whole core fuel cassette good for 20 years (at a capacity factor of 0.9) in a STAR power plant represents a 188 GW_{th} power transmission conduit. A single ship carrying ten cassettes on an itinerant one-month delivery voyage from a regional centre could supply nearly 1000 GW_{th} (1 terawatt years/year) to its service region. A fleet of 10 ships could provide 10 $terawatt_{thermal}$ years/year (which rivals the entire current world primary energy use of 12 $terrawatt_{thermal}$ years/year).

Marchetti has observed [3] that the economical scale of equipment sited at the “hubs” will expand to match the energy demand in the geographical area circumscribed by the spokes. Because of the enormous energy density of nuclear fuel contained in the refuelling cassettes, the “reach” of the nuclear fuel supply “spokes” through practical sized transport conduits (ships) (i.e., the energy demand met in the area circumscribed by the spokes) can be thousands of miles and as a result the fuel cycle facilities at the regional centres can (must) be sized for economy of scale to service the very large demand arising from a significant global region. Even if providing for a plausible world demand (~50 $terawatt_{thermal}$ years/year) by mid century, no more than a dozen such fuel cycle centres could meet the world’s *entire* primary energy needs. In that sense they could be viewed as the 21st century analogue to the oil fields of the twentieth century.

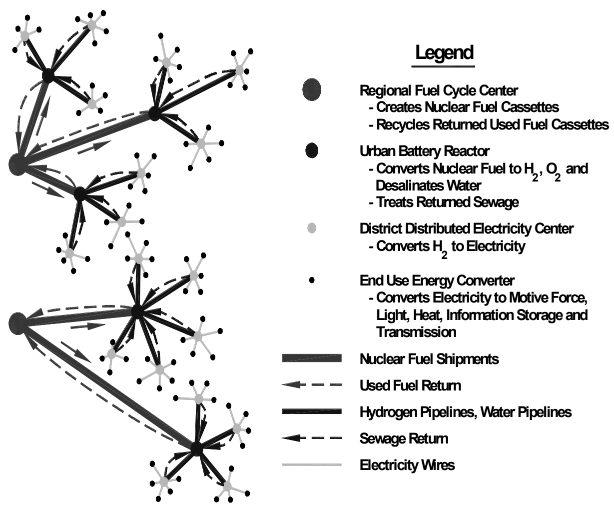

FIG. 1. Hierarchical hub/spoke energy architecture.

The "reach" of the next link in the supply chain – the hydrogen pipeline "spokes" – would reflect their several GW carrying capacity [4] and would service regions of several hundred mile dimension though pipeline grids such as are currently used to distribute natural gas to load centres. Pipeline grids would carry hydrogen (and water) to district centres scattered throughout the city and its surrounding population region. At a primary energy use rate of 4 toe/capita/year[3] (i.e., ~5.5 kW_{th} day/person day) a five and a half GW hydrogen pipeline could service a city and its environs with a population of a million people.

After manufacture at a STAR- H_2 power plant located at the margins of the city, the hydrogen and water will be piped or trucked to city districts through a grid of distribution conduits. At district level distribution hubs, the hydrogen will be partitioned to meet society's energy service needs:

- A third will be dispensed for hydrogen-fuelled transportation services;
- A third will be distributed by pipe throughout the district for heating homes, apartments, offices, and factories; and
- A third will be converted in fuel cells and/or micro turbines to electricity for distribution throughout the district.

The "reach" of the electricity distribution wires starting at district micro turbine or fuel cell converters of hydrogen to electricity and taking the electricity to final use in lighting, motors, and information management would be of the scale of city districts and skyscrapers – as is the current usage. This last stage of distribution would use the existing electrical and water distribution network (where it already exists) and *would thereby make the conversion to the new energy architecture nearly transparent to the end user of energy services.*

By mid century, district-level conversion of hydrogen to electricity – as opposed to conversion of heat to electricity at the STAR reactor sited at the city perimeter – is envisioned for several reasons. The first – and the one, which is already driving a transition – is supply reliability. Micro turbines and (imminently) fuel cells can provide secure electricity at a district level, even if the broader grid suffers a shutdown, because they run on a storable supply – currently natural gas, but eventually hydrogen. Some planners believe that distributed generators will, in fact, eventually *drive* the grid.

The second driver is that the hot water produced as the "waste" from conversion of hydrogen to electricity at district hubs can be used in support of the city's hot water needs. This will increase billable product for the owner of the conversion equipment but more importantly, it will reduce the water vapour and thermal plume ecological footprint of the conversion step. *This sets the scale of electricity production at a district level because of the limited "reach" of hot water distribution spokes.*

The overall conversion efficiency of nuclear heat to district-level re-conversion of hydrogen to electricity, [fission heat → hydrogen → electricity] would be about 0.45 x 0.80 = .36 which is already better than current LWRs. The overall conversion efficiency of nuclear heat to district level *energy products* [fission heat → hydrogen → electricity + hot water] would be about 0.45. When potable water manufacture from the STAR- H_2 process plant is included, the overall conversion [nuclear heat → energy services] reaches 85%.

The proposed hub/spoke energy architecture optimised for nuclear thus envisions a worldwide total of a dozen or less regional fuel cycle centres each servicing thousands of long refuelling

[3] 4 tons of oil equivalent per capita year (toe/capita year) is the average current primary energy use rate in Europe.

interval STAR battery heat source reactors which individually or in clusters service cities and their surrounding regions with hydrogen and potable water. Hydrogen substitutes for fossil fuel in the transportation and heating sectors. Electricity and hot water are produced from hydrogen at distributed district centres and electricity reaches its final point of use through wires.

Carrier conduit cross connections of the user hubs to multiple supplier hubs (not shown in Fig. 1) and *energy storage buffers* provided by the storable nuclear fuel and hydrogen energy carriers would provide for robustness of energy security at both the national and the individual user levels – and for protection against monopolistic pricing.

Over time, in a transition lasting of the order of a century, the hydrogen would gradually displace oil, gas, and coal and the new sustainable, nuclear-based architecture would gradually replace the current fossil-based world energy supply infrastructure. The resulting fission based energy supply architecture will provide centuries of energy on the known plus speculative ore base recoverable at $\leq$ \$130/kg U.

4. BREAKING THE ENERGY SECURITY/NONPROLIFERATION DILEMMA

Long, 20-year refuelling interval and full core cassette refuelling supported from consortia-owned regional fuel cycle (front and back end including waste management) service centres, operating under international oversight are intended to make nuclear-based energy supply available in countries that don't wish to emplace an indigenous front-to-back fuel cycle infrastructure – and to do so without jeopardizing their energy security posture. Consortia of client nations exercising ownership and control of the centre under international law – when combined with 20-year fuel cassettes installed on sovereign territory – could provide high assurances for a nation's energy security. At the same time, the regional centres, infrequent cassette refuelling and full transuranic recycle (such that both reload and spent fuel cassettes meet the spent fuel standard of self protection and no fissile material ends up in waste) are intended to provide appropriate barriers to misuse of materials and facilities for military purposes.

All fuel cassette shipments, refuelling operations and used cassette returns are conducted by regional centre personnel who bring the refuelling equipment with them and take it away with the spent cassette. No refuelling equipment remains at the site. The cassette and the cask are massive (each ~200 tonnes)[4] and are amenable to GPS monitoring for item accountancy providing resistance to diversion. Shipping is done with the cassette entombed in frozen Pb as a precaution against loss at sea. A whole core cassette refuelling operation using relocatable regional centre equipment is illustrated in Fig. 2.

This proposed architecture meets the criteria recently specified by M. ElBaradei for an enhanced non-proliferation regime [5].

5. SUISTAINABILITY AND ECOLOGICAL COMPATIBILITY

Full trans-uranic multi-recycle is employed to extract the full energy content from the uranium ore, and to consign only fission products (and trace recycle/refabrication losses) to waste. *A millennium of global energy supply can be supported by the currently known plus speculative uranium resource base recoverable at <130 \$/kg.*

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[4] Note that crawler cranes in use for LWR construction have capacities in excess of 600 tonnes.

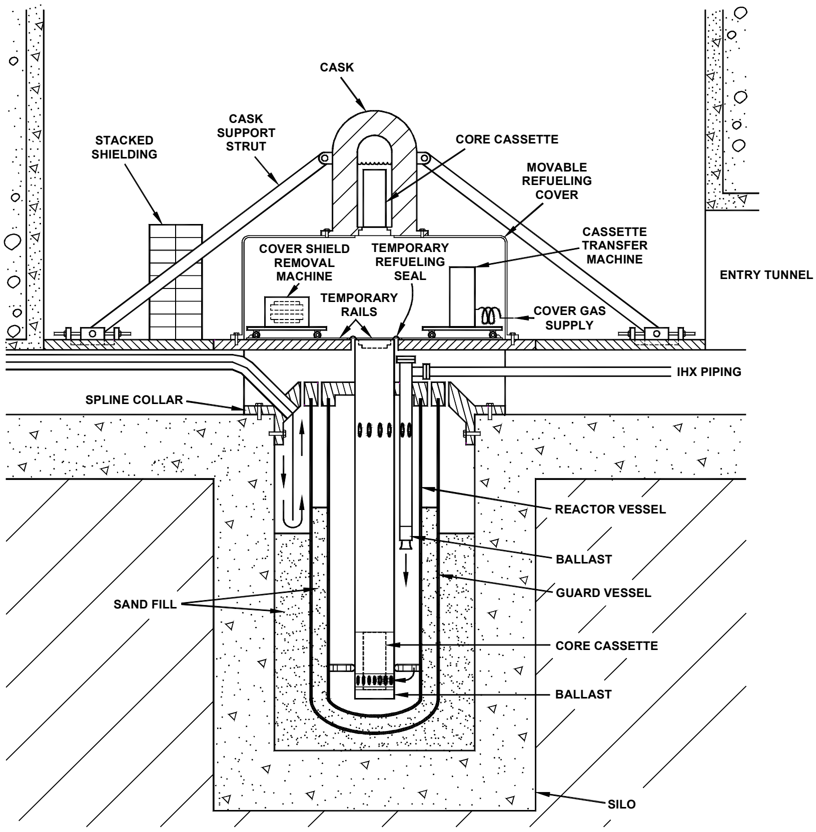


Fig. 2. Whole core cassette refuelling using relocatable equipment from the regional centre.

The fuel cycle feedstock is natural or depleted uranium, and multi recycle through sequential cassette reload cycles achieves total fission consumption of the feedstock; only fission product waste forms (and trace losses of transuranium nuclides) go to a geologic repository operated by the regional centre. These waste forms – lacking any transuranic component – decay to the equivalent radio toxicity levels of the original ore within 200-300 years

The architecture employs non-carbon emitting technology throughout the entire energy supply chain. With nuclear generation of hydrogen and a closed fuel cycle, this architecture achieves sustainable ecological closure – no carbon emissions anywhere in the supply chain; recycle of oxygen and water via nature's cycles; and self-consumption of long-lived radiotoxic isotopes such that only fission products go to waste. Radioactive waste stewardship is reduced from thousands to a few hundred years at which time neutral radiotoxic exchange is attained between ore withdrawals and fission product waste return to the earth's crust.

If successfully deployed such a nuclear-based global energy architecture can meet all elements of the broad definition of sustainability for global energy supply [2.]

6. STAR FUEL CYCLE TECHNOLOGY

The STAR reactors use uranium/transuranic N-15 enriched nitride fuel and operate on a 20-year whole core cassette-refuelling interval; they are fissile self sufficient with an internal core conversion ratio of one. The fuel recycle technology will be based on electrometallurgical recycle and remote vibropack refabrication of the uranium/transuranic nitride fuel. The recycle technology produces fuel feedstock comprising a commixed stream of all transuranium nuclides and achieves incomplete fission product removal such that the transuranic materials during processing and during fresh and used cassette shipping are always at least as unattractive for military use as is LWR spent fuel. The waste forms contain only fission products (and trace recycle losses of transuranium nuclides).

7. THE STAR HEAT SOURCE REACTORS

Table 1 summarizes the portfolio of STAR power plant concepts under current development by a team of Argonne, Livermore, and Los Alamos National Laboratories. STAR-LM and STAR-H_2 are 400 MW_{th} Pb-cooled, fast neutron spectrum, reactors operating at a power density similar to that of LWRs. Average discharge burn-up near 100 MW_{th}d/kg is achieved. They employ natural circulation cooling at full power and passive load following and passive safety response characteristics. The 400 MW_{th} sizing is optimised as high as possible while retaining a rail shippable reactor vessel size while allowing for natural circulation heat removal. The neutronics properties of lead coolant enable a high coolant volume fraction fuel pin lattice so that natural circulation will remove the heat at full power. Its neutron reflection properties and hard neutron spectrum permit fissile self regeneration in the core lattice itself which achieves zero burn-up reactivity loss over the 20-year burn-up interval and minimal reactivity vested in control rods – which is the key to enable passive load following/passive safety.

TABLE 1 STAR PORTFOLIO – SUPPORTING A MULTI-DECADE EVOLUTIONARY TRANSITION

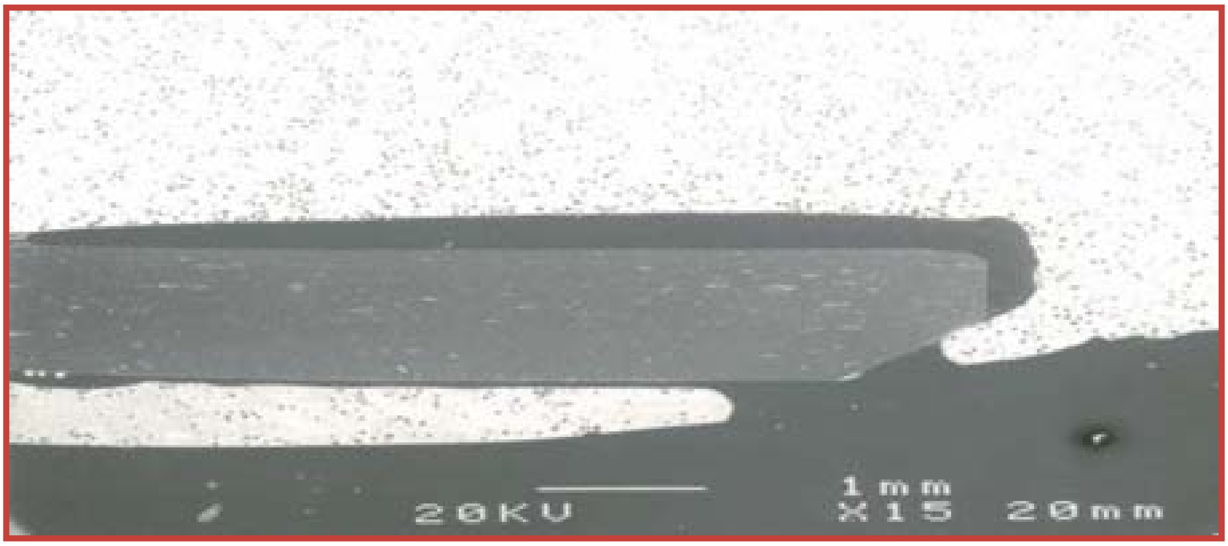
Portfolio Member	Power	Coolant	Tout	Converter	Products	Client*	Deployment Target
SSTAR S=Small	20-50 MW_{th}	Pb natural circulat.	550°C	rankine steam - or SC-CO_2 Brayton ↓ desalination	electricity + potable water or potable water	electri city for remote town of approx. 6,500	~2015 potential 1st prototype
STAR-**LM** LM= Liquid Metal	400 MW_{th}	Pb natural circulat.	550°C – 580°C	SC-CO_2 Brayton ↓ desalination	electricity +potable water	electri city for city of approx. 115,000	~2020
STAR-**H_2** H_2= Hydro gen	400 MW_{th}	Pb natural circulat.	800°C	Ca-Br thermo chemical cycle ↓ SC-CO_2 Brayton ↓ desalination	H_2 + potable water	all primay ry energy and potable water for city of approx. 25,000	~2030

* Assume 4 toe/capita year primary energy ≡ 5.3 kW $_{th}$ year/person year
Assumes 1/3 of primary energy converted to electricity

Passive safety/passive load follow in turn enables use of a balance of plant having no nuclear safety function; *this allows for indigenous construction and operation of the Balance of Plant, using the local work force and institutional conditions prevalent during the initial stages of industrialization, thus providing local jobs for economic growth.*

The difference between STAR-LM and STAR- H_2 is in core outlet temperature (565 vs. 800°C respectively). STAR-LM uses relatively conventional structural materials (ferritic martensitic steels) and Russian-developed coolant chemistry control technology – facilitating readiness for market sooner than STAR- H_2 whose high outlet temperature will require an extended development period for qualifying high temperature structural materials. STAR-LM will drive a SC-CO_2 Brayton cycle with a desalination bottoming cycle while the higher outlet temperature of STAR- H_2 is intended to drive a Ca-Br thermochemical water cracking cycle followed by a SC-CO_2 Brayton cycle and a desalination bottoming cycle.

All STAR reactors drive a Brayton cycle rather than a Rankine steam cycle. Absence of a feedwater heater failure hazard means that the return temperature from the BOP to the in-vessel reactor heat exchangers is guaranteed to exceed the 327°C freezing temperature of Pb by at least 100°C – even in BOP upset conditions. This assurance combined with the less corrosive nature of Pb compared to Pb-Bi alloy as core outlet temperature is raised to 800°C – plus the avoidance of a Po source term have in combination led to our selection of Pb over Pb-Bi eutectic for STAR reactors. (Preliminary screening has shown encouraging performance of SiC or SiC composite for cladding STAR- H_2 and structures – see Fig. 3).



No evidence of lead interaction with SiC sample.

FIG. 3. Material screening for service in Pb – silicon carbide results: 1000 hrs in 800°C Pb at low O_2 content.

Mid-sizing of the STAR-LM and STAR- H_2 power plants is targeted for incremental deployments in support of urban centres of developing countries when capital financing is dear and/or indigenous infrastructure is at an early stage of development. Modular construction, factory fabrication, and delivery of a turnkey heat source reactor to the client's site *where a non safety grade balance of plant has already been emplaced* will facilitate rapid assembly and initiation of revenue generation – strategies intended to achieve economy of mass production to replace historical economy of scale.

SSTAR is a downsized version of STAR-LM, which is intended for support of the energy security needs of remote, small villages. Its small size makes it a candidate for a first STAR concept prototype, which will exercise the institutional innovations (such as licensed design certification by test; factory fabrication/rapid site assembly; licensing of passive load follow, passive safety and non-conventional containment, etc.) upon which all STAR concept reactors rely.

8. THE STAR-LM AND SSTAR SUPERCRITICAL CO2 BRAYTON CYCLE BALANCE OF PLANT

A supercritical CO_2 (SC-CO_2) Brayton cycle is being developed for the STAR concepts [9]. The critical point for CO_2 is at about 31°C and 7 MPa pressure. The Feher cycle design we are considering operates from just above the critical point to ~600°C and 20 MPa at the input to the turbine. Two features of this cycle are extremely attractive; first, a heat to electricity conversion efficiency near 45% can be attained at only 550 to 600°C turbine inlet temperature – owing to significantly reduced compression work just above the critical point where SC-CO_2 density is dramatically higher than that of ideal gases such as He (see Table 2). Second, the rotating machinery is significantly smaller than for He equipment and is massively smaller than for a steam cycle of the same rating (see Table 3).

TABLE 2. COMPARISON OF DENSITIES

Fluid	Location	Pressure, MPa	Temperature, °C	Density, kg/m^3
S-CO_2 (STAR-LM)	critical point	7.37	30.98	468
	cooler outlet	7.40	32.15	369
	compressor outlet	20.0	84.0	568
	turbine inlet	19.85	550	124
	turbine outlet	7.44	425	56.3
Helium (Eskom PBMR)	cooler outlet/ compressor inlet	2.6	27	4.17
	compressor outlet	7.0	104	8.93
Water		0.1	20	998
Lead		0.1	495	10,400
Sodium		0.1	420	828

TABLE 3. RESULTS OF SC-CO_2 TURBINE AND COMPRESSOR DESIGN ANALYSES FOR A 400 MWTH STAR-LM BRAYTON CYCLE

	Turbine	Compressor 1	Compressor 2
Number of stages	4	4	4
Length (without casing), m	0.8	0.5	0.3
Maximum diameter (without casing), m	1.25	0.5	0.7
Efficiency without secondary losses, %	95.9	96.0	95.3
Assumed secondary losses, %	5	5	5
Net efficiency, %	90.9	91.0	90.3

The very strong increase of SC-CO_2 thermal capacity, Cp, near the critical point necessitates a partitioning of the recuperator into high temperature and low temperature segments such that only a fraction of the low temperature (high Cp) compressed CO_2 is used to cool the full flow of turbine exhaust in the low temperature recuperator; the matching of heat losses and gains by partitioning the flow through low and high temperature recuperator raises the overall (HTR + LTR) recuperator efficiency to above 90%.

Figure 4 illustrates the reactor and BOP thermal conditions for the STAR-LM plant design point. It achieves a 44% conversion of heat to electricity – producing 175 MW_e.

A desalination bottoming cycle will fit conveniently under the low temperature recuperator – replacing the Brayton cycle cooler. Such a bottoming cycle is illustrated next for STAR- H_2.

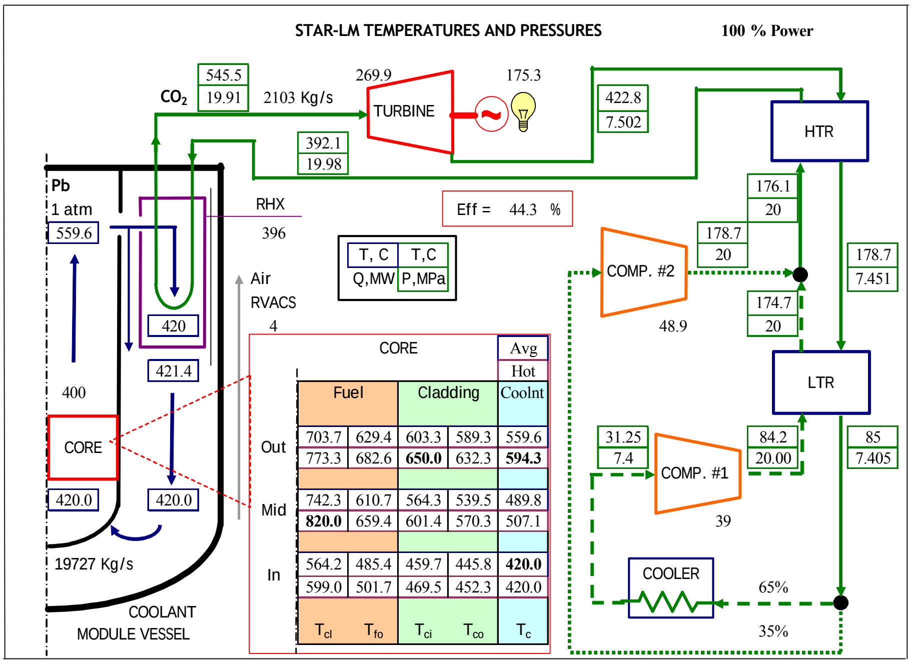


FIG. 4. Thermal hydraulic conditions and heat balance for STAR-LM coupled to S-CO_2 Brayton cycle.

9. THE STAR- H_2 BALANCE OF PLANT

The STAR- H_2 balance of plant (BOP) is comprised of three cascaded cycles (water cracking; Brayton cycle; desalination) operating at successively lower temperatures – and with the heat rejected from each cycle used to drive the succeeding cycle (See Fig. 5) [6]. The reactor supplies 400 MW_{th} of heat between 800ºC and ~650ºC to the BOP through an ambient pressure flibe (fused salt) intermediate loop. The strategy for BOP plant design is to use as much of the heat as possible to maximize hydrogen production; use only as much heat to make electricity in the Brayton cycle as is required to run the BOP; and use whatever heat is finally left over to desalinate water. Converting as much as possible of the low grade heat to potable water minimizes the ecological thermal footprint, and the brine tailings are rejected only slightly above ambient seawater temperature.

The Ca-Br water cracking cycle [8] has three main segments: An endothermic “water cracking” segment where $CaBr_2$ and steam react at 750°C to make HBr and CaO; an exothermic Ca re-bromination segment where CaO and bromine react at 600°C to regenerate $CaBr_2$ for recycle and release heat and oxygen; and a plasma chemistry HBr cracking segment where electrical driven (RF frequency) energy cracks HBr at 90°C to regenerate bromine for recycle and to release hydrogen. The plasmatron is followed by a pressure swing absorption cascade, which cleans and pressurizes the hydrogen to meet pipeline delivery specifications.

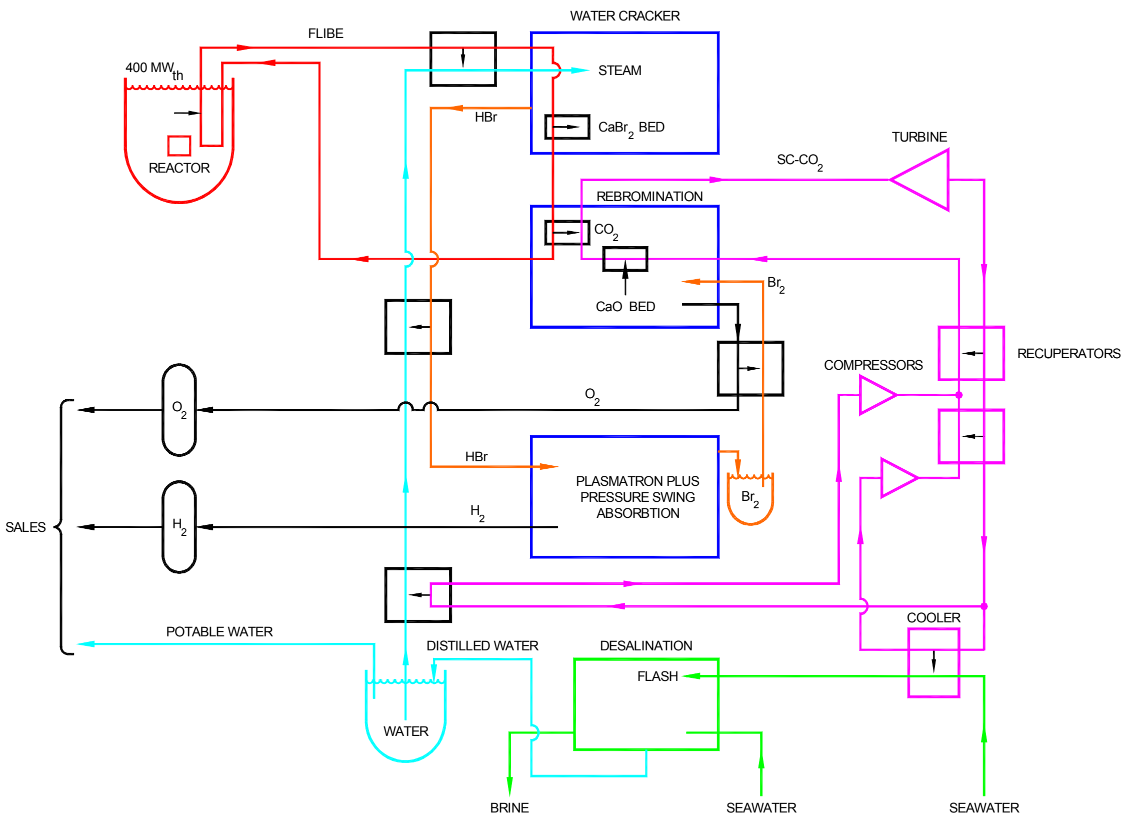


FIG. 5. Process heat cascade: water cracking → Brayton cycle → desalination.

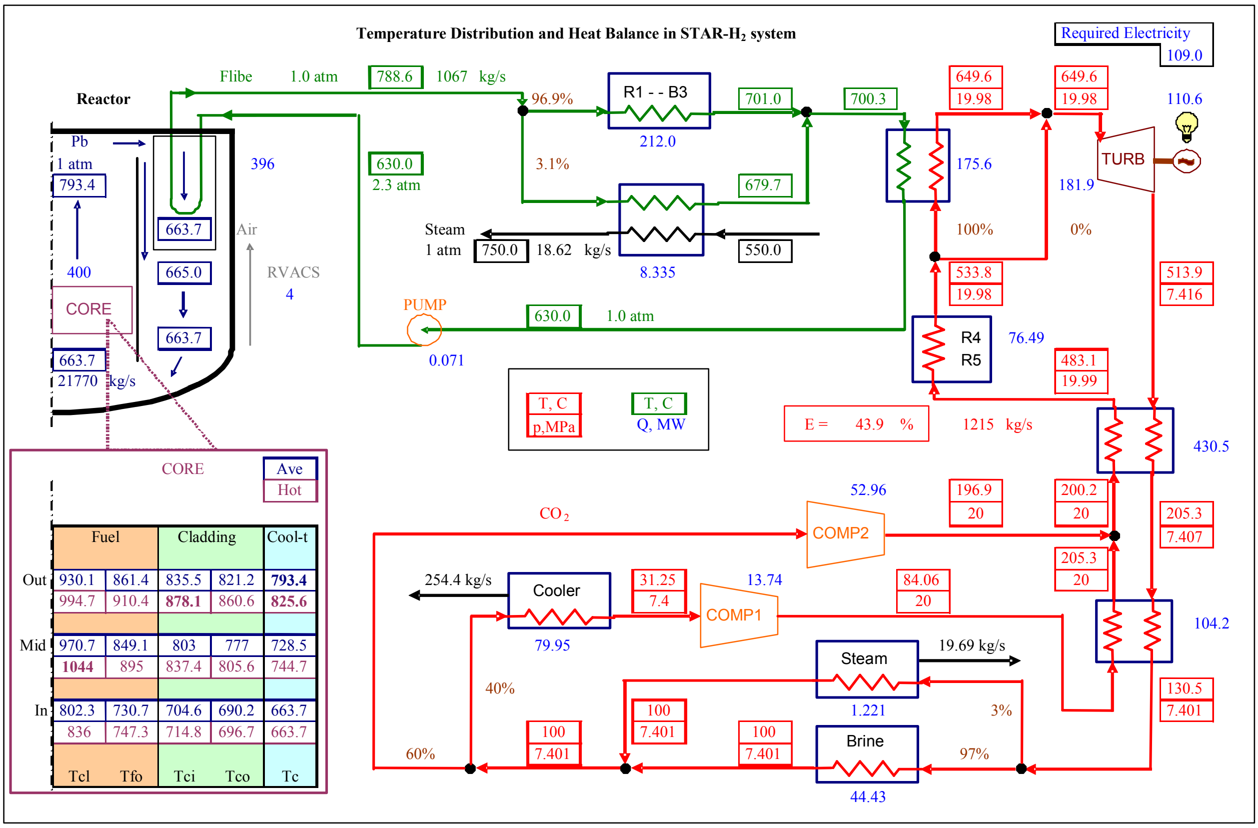


FIG. 6. STAR-H₂ plant configuration at 100% load: water cracking → Brayton cycle → desalination.

Heat at 600ºC is rejected in the segment for regeneration of $CaBr_2$ from CaO. It is used to help drive the SC-CO_2 Brayton cycle. After expansion in the Brayton cycle turbine, the SC-CO_2 passes through a high temperature and then a low temperature recuperator. It exits the low temperature recuperator at 125ºC. Heat is rejected from the SC-CO_2 to seawater in the Brayton cycle cooler, which cools the SC-CO_2 from 125°C to 31ºC in preparation for its compression.

The 100ºC seawater exiting from the Brayton cycle cooler delivers heat and seawater feedstock to the desalination plant. The desalination plant is a feed forward Multi-Effect-Distillation (MED) design [10], which produces 8000 m^3/d of potable water. Finally, heat at temperature slightly above ambient exits the plant in the form of heated brine tailings from the desalination process – minimizing the thermal plume ecological footprint of the plant. Alternate bottoming cycles have been identified for use at landlocked sites.

Figure 6 illustrates the reactor and BOP thermal conditions for the STAR- H_2 design point.

The STAR- H_2 BOP has been designed at the conceptual level and shown to achieve about 44% conversion of heat to H_2 (LHV) – making 160 MW_{th} days/day of H_2 (LHV) and 8000 m^3/day of water – enough to support all primary energy and water needs for a city of 25,000 using energy at 4 toe/capita year and water at 300 litres/day/person. All the electricity produced by the Brayton Cycle is consumed on site (and is figured into the 44% heat to H_2 conversion). Overall, 85% of the reactor's 400 MW_{th} is converted to energy products; 15% is rejected in the form of heated brine.

10. PASSIVE SAFETY PERFORMANCE

The STAR reactor concepts employ an ambient pressure primary and extensive levels of passive safety [7, 13] to be consistent with a worldwide deployment of many thousands of STAR plants; to remove all nuclear safety functions from the balance of plant; and to facilitate siting near urban centres.

The passive safety performance has numerous facets. First, coolant flow is by natural circulation even at full power. Moreover, core power/flow ratio is maintained in a safe range by innate reactivity feedbacks such that Anticipated Transients Without Scram (Station Blackout, Loss of Heat Sink, etc.) are all safely accommodated by passive means. Design for internal conversion ratio of unity – yielding less than a dollar of burn-up swing – eliminates rod run-out event hazards; ambient pressure primary and use of a top entry reactor vessel with a backup guard vessel eliminates the loss of coolant hazard. Chemical compatibility of fuel, coolant, and cladding and an open ductless pin lattice allows for run beyond clad breach absent the creation of sludges, blockages, and explosive gases. An always-operating natural draft decay heat transport channel from the guard vessel exterior to ambient plus a large thermal inertia in the coolant pool facilitates passive decay heat removal. These factors innately terminate accident progression pathways prior to core disruption and thereby allow for use of a high surface/volume guard vessel containment design and a balance of plant having no nuclear safety function. This facilitates construction of the BOP to local standards using local labour creating local jobs to assist economic development.

The reactor is located in a silo under an earthen berk to protect it from external hazards (including – in the case of STAR-H_2 – those posed by the co-sited hydrogen production plant) (See Fig. 7).

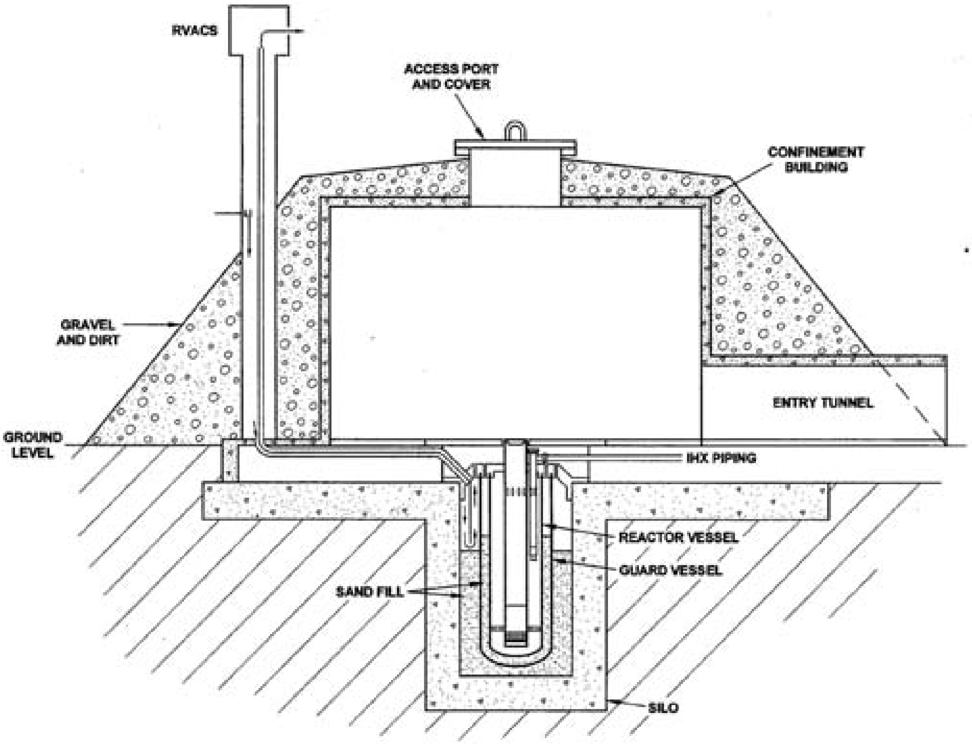


FIG. 7. STAR- H_2 heat source reactor sited in a silo.

11. PASSIVE LOAD FOLLOW PERFORMANCE

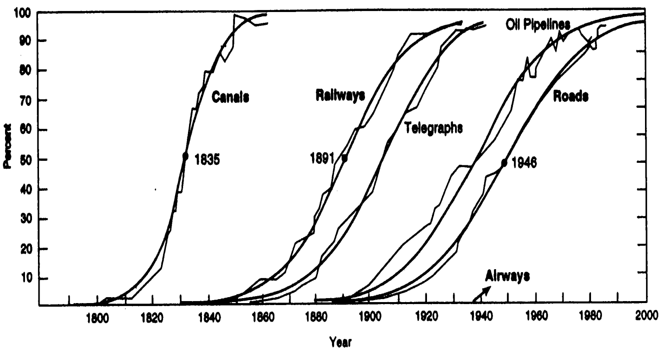
The STAR power plants are designed for passive load follow wherein passive reactivity feedbacks innately adjust the core power to match the heat request from the balance of plant. The heat request is communicated via flow rate and return temperature of the heat transport loop connecting the BOP to the reactor – i.e., not by an active control system. This passive load follow strategy maintains power/flow ratio in a safe range *no matter what request (purposeful or spurious) is communicated to the reactor.* This feature basically precludes any single or combination of equipment and/or human failure events in the BOP from causing reactor damage – and it not only simplifies the plant design but also allows to reduce the staffing level and skill level of the BOP operating and maintenance crews.

Conceptual designs of the STAR-LM and STAR- H_2 power plants have progressed to the stage of having shown that they can meet their long life, natural circulation cooling, and passive safety/passive load follow requirements.

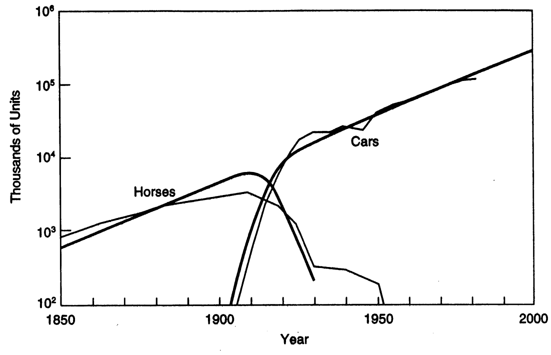
12. INCREMENTAL MARKET PENETRATION BY MEETING CLIENT NEEDS

The STAR hierarchical hub-spoke architecture is a world energy supply architecture optimised for nuclear and intended to displace the current fossil-based energy architecture over a multi decade market substitution transition period characteristic of any major infrastructure transition; (e.g. the pace of infrastructure substitutions for the US transportation sector shown in Fig. 8 illustrates the ponderous dynamics of market substitutions of massive infrastructures).

Figure 8 illustrates a transition period of four or five decades. The new technology displaces the old incrementally only as former long-lived assets have to be replaced and in the face of resistance by the entrenched enterprise who employ both institutional and technological efforts to maintain market share.



Growth of infrastructures in the United States as a percentage of their maximum network size. SOURCE: Grübler and Nakićenović (1991).



Number of non-farm draft animals and automobiles. SOURCE: Nakićenović (1986).

FIG. 8. Characteristic 40-50 year incremental market penetration dynamics of major infrastructures.

So what causes a market substitution to take place? Contemplation of Figure 8 for history of the transportation sector reveals the underlying drivers for market substitutions; -- they come down basically to the new technology meeting client needs for (unchanging) services in a "better" way. "Better" encompasses many considerations of which cost is only one; for example railroads were both faster and finer-grained in geographic coverage for moving people and goods than were canals – and roads were even more so. Automobiles replaced horses not because the resource base for horses had diminished and their price increased but because automobiles were simply more convenient and cleaner. Similarly in the energy field, petrochemicals displaced coal for transportation services not for lack of coal resources but because of versatility and convenience. Coal still supplies 60% of US electricity.

These lessons suggest that success for STAR concept market penetration – if it happens – will occur primarily because of its having been tailored specifically to the energy service needs and financial and institutional conditions of the targeted clients – the cities of developing countries.

Incremental market entry is facilitated because the proposed hierarchical hub-spoke nuclear assets – both reactor plants and fuel cycle facilities – are scalable and because the end user of energy services will find the substitution nearly transparent; electricity, water, and chemical fuel (hydrogen) will continue to be produced – from a different source. Being scaleable, the infrastructure need not be deployed all at once; clusters of power plants can be built up incrementally at a city's perimeter as population and/or energy use/capita increase. Additional process lines can be emplaced incrementally at regional fuel cycle centres to increase mass throughput rates. Moreover, STAR- H_2 will be preceded in deployment by sister concepts in the STAR family such as SSTAR and STAR-LM since in most respects all STAR plant concepts are similar *and they all rely on the same institutional arrangements for regional fuel cycle centres and on the same business strategies.* Since STAR-LM and SSTAR are less aggressive in core outlet temperature – using traditional structural materials and producing electricity from a SC-CO_2 Brayton cycle operating at 565°C; they will be ready for market earlier than STAR- H_2 and the existence of a full-fledged hydrogen economy, but their deployments will initiate the transition to the regional fuel cycle centre infrastructure and new business model and institutional arrangements.

13. MANAGING THE TRANSITION AND FUELING A MILLENIUM OF SUSTAINABLE DEVELOPMENT VIA SYMPIOTIC FUEL CYCLES

The emplacement of a dozen or less regional fuel cycle centres will provide the mechanism to manage the world's inventories of fissile material and the nuclear waste produced throughout a transition involving coexistence of a symbiotic mix of reactor types serving different market segments. Clearly, open cycle LWRs will maintain a growing and significant global nuclear market share for decades. Working inventories for new STAR deployments will initially come from reprocessing LWR spent fuel. This symbiotic fuel cycle allows for incremental market penetration of STARs by beneficially managing the "waste" from the current once-through LWR cycle (such that only fission products go to a repository) while simultaneously beneficially providing fissile transuranic feedstock for initial working inventories of STAR deployments. Since the distributed STAR heat source reactors are fissile self sufficient, once started up their refuelling cassette refurbishment requires only U238 feedstock and each STAR and its replacements would maintain a steady energy supply for many centuries while fed only by U238. However, for fuelling a *growing* deployment of STARs, the LWR source of feedstock will not provide for more than several decades of growth until the economically recoverable uranium ore reserves driving the LWR open cycle are depleted [14]. *Sustained growth will ultimately require the presence of fast breeder reactors.*

In the architecture proposed here, fast breeder reactors will ultimately be sited at the regional fuel cycle centres to manufacture excess fissile material to fuel new deployments in a growing economy after the source of fissile from LWR spent fuel becomes exhausted. The heat from their operation will be converted to hydrogen for shipment to regional consumers.

At the end of the transition period a sustainable, growing, fissile self-generating nuclear energy architecture will be driven by the world's U238 resource base. The energy shares of battery plants vs. breeders in the enterprise will satisfy a simple fissile balance equation wherein the fissile generation rate *integrated over all reactors in the complex* must be sufficient to refuel the existing fleet and also to produce an excess sufficient to provide the working inventories for the next round of new deployments.

Consider an energy enterprise growing at an asymptotic rate, α [%/year] comprised of two reactor types – batteries and breeders – with energy delivery fraction, F, at distributed battery reactors and energy delivery fraction (1-F) at centralized breeder reactors. Suppose that the battery reactors require a transuranic working inventory (including out of reactor recycle lag time) of I_s [kg/MW_{th} heat rating] and operate at a transuranic conversion ratio of unity, $CR = \frac{kg\,TRU\ produced}{kg\,TRU\ burned} \equiv 1.0$, and that the corresponding attributes for breeders are I_B and $(BR-1) = \left[\frac{kg\,TRU\ produced - kg\,TRU\ burned}{kg\,TRU\ burned}\right]$. Then, what is the architecture's distributed energy fraction, F, of battery reactors as a function of α, I_S, CR, I_B and BR?

At any given time with the nuclear architecture delivering P fissions/year[5]

$$\frac{\text{Net TRU mass gain}}{\text{year}} = \{F * CR + (1-F)BR - 1\}P \tag{1}$$

$$\text{TRU mass needed to double capacity} = \{F * I_S + (1-F)I_B\}P \tag{2}$$

[5] For this scoping analysis, differences in capacity factor are ignored.

Time, τ, needed to generate the fissile transuranium nuclides inventory required to double deployed capacity

$$= \frac{\{F * I_S + (1-F) I_B\} P}{\{F * CR + (1-F) BR - 1\} P} \tag{3}$$

Set τ to doubling demand

$$\tau = \frac{.693}{\alpha} = \tau_D \tag{4}$$

Put (4) in (3) and solve for F using CR=1

$$F = \frac{\frac{BR-1}{I_B}\tau_D - 1}{\frac{BR-1}{I_B}\tau_D - 1 + \frac{I_S}{I_B}} \tag{5}$$

If α is zero (no growth and $\tau_D = \infty$), then no breeders are needed, F=1, and Batteries alone are fully self-sustaining.

Note that $BR\text{-}1/I_B$ is the fraction of one breeder working inventory generated per year and when multiplied by demand doubling time, τ_D it is the fraction of one breeder working inventory generated over a world energy demand doubling time. If this product is less than one, nuclear can't keep up with demand – even if the whole complex is comprised of breeders.

But, if in one energy demand doubling time, a breeder can generate more than enough transuranium nuclides to build another breeder, then some fissile mass will be left over to build more Batteries, and the fraction, F, of Batteries in the architecture will be 0<F<1.

Equation 5 shows that the fraction of distributed energy production in Batteries, F, becomes larger:

- the slower is the energy demand growth rate;
- the shorter is the breeder doubling time; and
- the smaller is the battery fissile working inventory compared to the breeder working inventory.

Sodium cooled breeder reactors fuelled with *oxide* and using five year recycle periods can achieve 25 to 30 year doubling times and have total cycle working inventories, I_B, of no less than two in-core inventories and more likely about three. Sodium cooled, *metal alloy* fuelled breeders using pyro-metallurgical recycle of one or two year recycle periods [15] can achieve 10 to 20 year doubling times – with working inventory, I_B, of two in-core inventories because of short cooling times. (Gas and Pb cooled fast reactors have very much longer doubling times and are unsuitable for the function discussed here.)

The battery reactors achieve long refuelling interval by derating the fuel specific power compared to Na breeders – so that their in-core fissile inventory is 4 or 5 times that of a sodium cooled breeder. However the recycle lag time working inventory is only 1/20 to 1/10 of the in-core inventory – thus I_s/I_B is about 2:

Figure 9 shows the value of battery energy fraction, (F), in the energy enterprise vs. energy demand doubling time (or growth rate, % per annum) – parametric in breeder doubling time for $I_S/I_B = 2$. It is clear that:

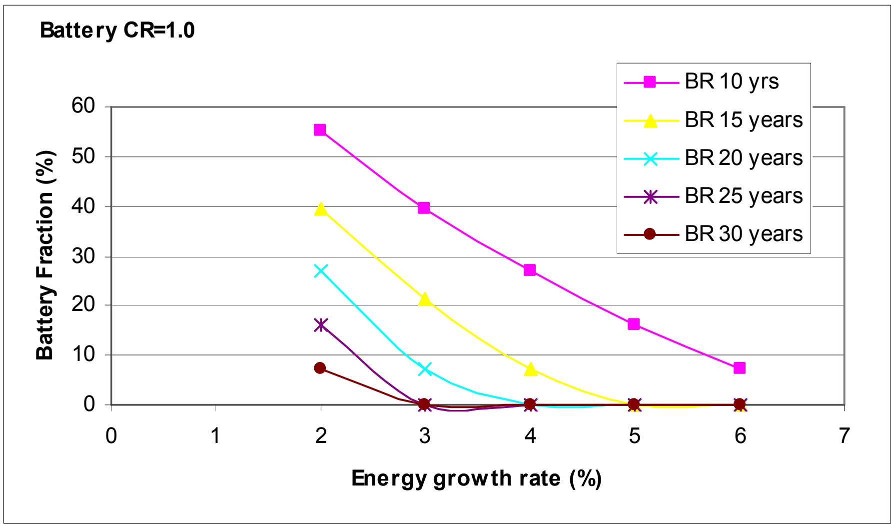


FIG. 9. Battery energy fraction as a function of demand growth rate, parametric in breeder doubling time (for $I_S/I_B=2$).

- A significant fraction (1-F) of the energy in a growing nuclear enterprise will have to be generated at the regional fuel cycle centres by the breeders;
- To increase the fraction of energy to be delivered by distributed battery, reactors, one should seek to decrease breeder-doubling time[6] and to increase battery energy output per refuelling interval (decrease I_S/I_B).

14. BUSINESS PLAN INNOVATIONS

A growing market can be foreseen for secure energy supply at moderate buy-in cost and with outsourced fuel cycle and waste management support; the STAR concept is designed to meet this need. *The nuclear industry will have to undergo significant structural changes in order to support an expanding STAR segment of nuclear energy supply;* the battery heat source reactor business will likely become one analogous to the airplane and automobile sales businesses where risk is transferred from client to supplier; where customers receive a commodity product delivered turnkey and ready to use; and where suppliers make significant upfront investments in factories and distribution logistics infrastructure to attain economy of mass production and to spread their investment cost over a large volume of sales.

The STAR whole core fuel cassettes contain a significant fissile loading – which if it had been employed instead in a conventional high power density fast reactor or in a low specific inventory thermal reactor would have generated revenue at a higher rate per kg of fissile material. The energy security attained via the long refuelling interval in STARs leads to an economic competitive disadvantage in fuel cost; this will require business plan innovations. As an example, since the STAR refuelling cassettes are fissile self sufficient, there is no loss

[6] If it turns out the breeder doubling time is too long, and the battery energy fraction becomes too small; more advanced breeding technologies might be placed at the regional fuel cycle centres. These might include high current proton accelerators driving spallation neutron targets to drive transuranic production in U238 blankets. Or neutrons from fusion devices might be used to produce copious neutron fluxes on U238 blankets.

of fissile mass "principal" over the burn cycle. The cassettes at the end of life contain the same fissile mass "principal" as at beginning of life. This is the character of a monetary bond; third party long term investors such as insurance companies or retirement fund managers who invest in "safe" bonds might consider to make a business of investing in cassettes and leasing them to power plant clients who would pay a monthly expense for use of the non-depleting fissile inventory in the cassette. This business innovation could potentially meet the needs of lessor who would not loose principal and lessee who would avoid upfront capital cost – and could resemble the current auto, truck, and airplane leasing businesses.

The regional fuel cycle centre business may provide a natural transition opportunity for multi-national petrochemical firms to remain in the fuel supply field even as the world's fossil reserves diminish and as customer preference switches to hydrogen as a chemical energy carrier. Their traditional multi-tens of billions of dollars of annual investment in oil and gas reserve development (see Fig. 10) is more than sufficient for the capital outlay required to establish a regional fuel cycle centre; – and one such centre represents an energy supply for capacity greater even than a major oil field. With the breeders sited at the regional centres, a multinational petrochemical firm's transition to a business of nuclear fuel refining and hydrogen production and distribution could benefit from corporate expertise and their sunk cost distribution assets.

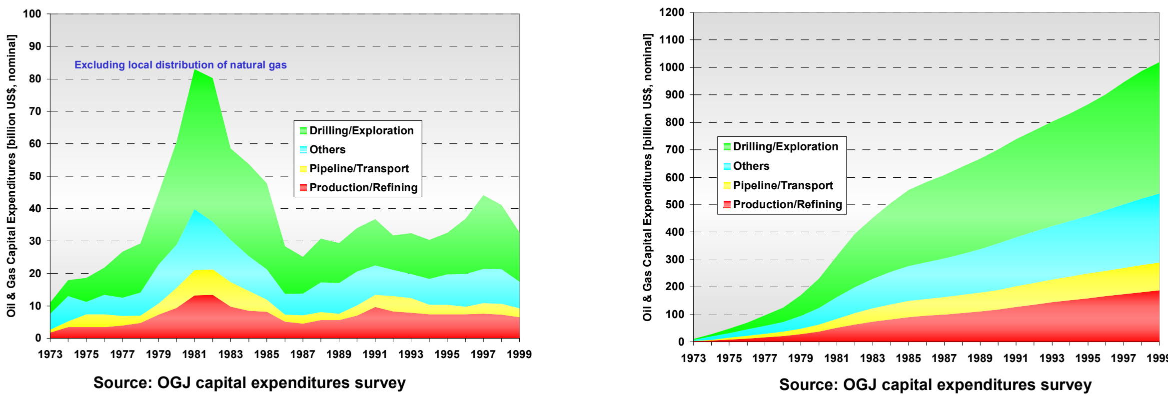


FIG. 10. Oil and gas industry investments in exploration, production and refining rose with price and environmental standards between 1973 and 1999 the U.S. oil and gas industry invested over $1.0 trillion (nominal).

15. INSTITUTIONAL INNOVATION

Two and a half centuries ago the switch from renewable to a stored chemical energy supply could not have fuelled the industrial revolution absent institutional changes to exploit the chemical fuel's higher energy density – by reorganizing the architecture of production (factories, division of labour, free markets, joint stock companies, etc. replaced the cottage industry, skilled labour guilds, a mercantile economy, and debt financing). Now, the proposed switch from chemical to even higher density nuclear energy supply to fuel a sustainable development revolution will also require institutional innovations to exploit nuclear's innate features.

Global deployment of STAR reactors meeting Gen-IV goals for sustainability, safety, energy security and non-proliferation rests on the use of regional fuel cycle centres owned by consortia of clients and operating under international non-proliferation oversight. A supplement to the NPT could be considered in which a nation agrees to forego emplacing an indigenous fuel cycle infrastructure in exchange for legally binding access to services from the regional centre.

National membership in the regional centre consortium and receipt of services for a nation's clients could be contingent on the nation's legal commitment to regulate its nuclear deployments in conformance with international norms on safety, radiological control, liability, mutual assistance and early notification of emergencies, non-proliferation treaty provisions, etc., etc. Supranational strategies for emplacement of regional fuel cycle centres and waste repositories on sovereign territory will be required – as for embassies, the UN, and EC headquarters, etc.

Additionally, licensing reciprocity agreements of STAR reactors between supplier and client regulatory authorities will have to be emplaced.

The emplacement of such a supranational legal regime for governing global nuclear energy will build on the substantial ensemble of legal norms already emplaced by the IAEA, EU, and NEA over previous years. However, full ratification and execution of all necessary provisions will require substantial further efforts in the international community.

16. SUMMARY

The outcome of the research on the STAR concept is a proposed nuclear-based energy supply architecture, which employs nuclear fuel, hydrogen and electricity as the energy carriers. Fuel is delivered in 20-year whole core refuelling cassettes to STAR plants placed near cities in both developed and developing countries. The STAR-LM plant is a small, (400 MW_{th}), long refuelling interval fast neutron spectrum reactor which provides fission heat to manufacture electricity for a developing country city of 110 000, while STAR- H_2, which will be deployed several decades later, manufactures hydrogen and potable water sufficient to meet the entire *primary* energy and water needs of a city of 25 000 in a developing country. STARs operate fissile self-sufficient; employ passive safety response; couple to a non-nuclear-safety-grade balance of plant and follow BOP heat requests using passive means. The STAR fuel cycle is based on full transuranic multi-recycle performed at regional fuel cycle centres – achieving a proliferation-resistant nuclear-based energy supply, which is sustainable for many centuries and is suitable for deployment worldwide. The regional fuel cycle centres also will eventually site breeder reactors whose function is to convert fertile U238 to fissile transuranium nuclides to fuel initial working inventories for a growing deployment of STARs. The fission heat from the breeders is converted to hydrogen for shipment to regional customers.

Over time the proposed nuclear-driven energy supply architecture would displace fossil and provide energy to support a global energy infrastructure meeting all aspects of sustainable development – secure longevity, ecological compatibility and social acceptability [2]. With concomitant institutional innovation it might succeed to fuel an increase in GDP/capita for the 80% of humanity, which has not yet been reached by the Industrial revolution.

ACKNOWLEDGEMENTS

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FIXED BED NUCLEAR REACTOR CONCEPT

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Abstract. Small nuclear reactors without the need for on-site refuelling have greater simplicity, better compliance with passive safety systems, and are more adequate for countries with small electricity grids and limited investment capabilities. The Fixed Bed Nuclear Reactor (FBNR) is based on Pressurized Water Reactor (PWR) technology. FBNR has an integrated primary circuit and is a small and simple, modular, inherently safe and passively cooled reactor with reduced adverse environmental impacts. Spherical fuel elements are fixed in a suspended core by the flow of water. Any accident will signal cutting off power to the coolant pump causing a stop in the flow of coolant. This will make the fuel elements fall out of the reactor core, driven by force of gravity, and enter the passively cooled fuel chamber where they would be stored in a safe sub-critical state. The simplicity and passive safety characteristics of FBNR together with the reliance on a well-established PWR technology make it a viable option for the near future deployment.

1. INTRODUCTION

Fixed Bed Nuclear Reactor (FBNR) [1,2] is a simplified version of the fluidized bed nuclear reactor [3-14]. In FBNR spherical fuel elements are at a fixed position within the core, therefore there is no concern about the consequences of friction between them, as often raised in relation to fluidized bed concept. In the latter case there is a need to study the degree of erosion in order to determine the required clad thickness. There is little work done on the fixed bed nuclear reactor concept so far, but the experience gained from the fluidized bed reactor design could essentially facilitate the development of FBNR.

2. REACTOR DESCRIPTION AND APPLICATIONS

FBNR is based on PWR technology. The reactor is modular in design and each module is factory fabricated and fuelled. The fuelled modules in sealed form are transported to and from the site. FBNR has long fuel cycle and operates without on-site refuelling.

FBNR has an integrated primary system design and allows for an incremental capacity increase through modular approach. The basic module has a reactor core and a steam generator in its upper part and a fuel chamber in its lower part. The core consists of a 25-cm diameter zircaloy tube in which, in the reactor operation, the spherical fuel elements are held together by coolant flow in a fixed bed configuration forming a suspended core. The fuel chamber is a 10-cm diameter tube made of an alloy with high neutron absorption that is located directly below the core. A steam generator of the shell-and-tube type is integrated into the upper part of the module. A thin neutron absorber shell slides inside the core tube, acting similar to a control rod. The reactor is provided with a pressurizer system to keep the coolant at a constant pressure. Each module has an independent pump. A crown type header on top of the reactor connects all modules into a unique system with integrated incoming and outgoing fluid flows.

Figures 1, 2, 3, and 4 illustrate the operation of FBNR module. The pump circulates the coolant inside the module moving it up through the fuel chamber, the core, and the steam generator and thereafter the coolant flows down back to the pump through a concentric annular passage, Fig. 1. At a certain pump velocity, the water coolant carries up the 8 mm diameter spherical fuel elements from the fuel chamber into the core. A fixed suspended core is formed in the upper part of the module, Fig. 2. In a shut down condition, the suspended core breaks down and the fuel elements are back in the fuel chamber, Fig. 4.

Any signal from any detector due to any type of accident is assumed to cut off power from the pump, causing the fuel elements leave the core and, driven by force of gravity, go back into the fuel chamber, see Fig. 3, where they remain in a highly sub-critical and passively cooled state, Fig. 4. The fuel chamber is cooled by natural convection that transfers heat to the surrounding air or to a pool of water.

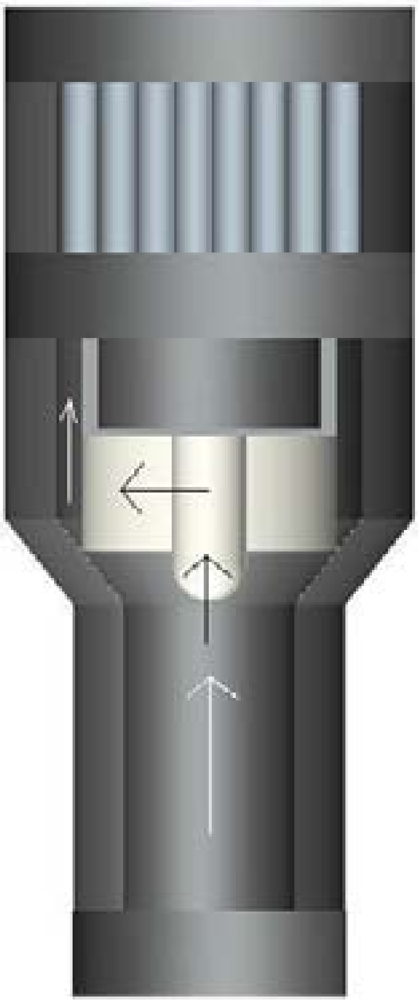

FIG. 1. Coolant flow path in FBNR module.

FIG. 2. Suspended core in operating condition.

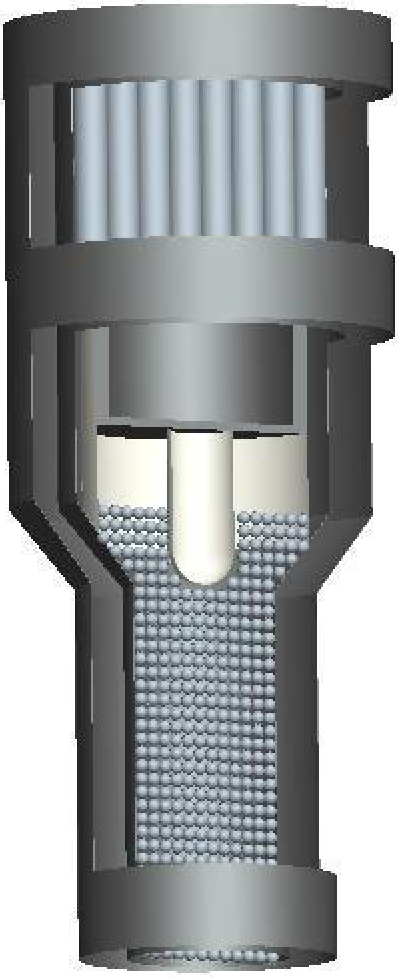

FIG. 3. Suspended core after
cut-off of the pumping power..

FIG. 4. Suspended core
in a safe shutdown state.

A detailed heat transfer analysis performed for the fuel elements has shown that due to a high convective heat transfer coefficient and also due to a large heat transfer surface-to-volume ratio, maximum power extracted from the reactor core is limited by the mass flow of coolant corresponding to the allowed pumping power ratio rather than by temperature limits of the core materials. The core tube is designed to be of a slightly conical shape in order to decrease the coolant velocity along core height and, through a compacting effect, to secure a more stable fixed bed.

The proposed reactor concept is very flexible in its nature, which makes it possible to consider several design options, such as indicated below.

2.1. Fixed bed with supercritical steam as coolant

The concept of a direct cycle reactor operating at supercritical pressure is attractive for improving the thermal efficiency, which would particularly lead to the reduction of adverse environmental impacts. The reactor combines the fixed bed concept with the idea of a direct cycle reactor operating at supercritical pressure [12]. The supercritical steam is used as reactor coolant. The critical pressure of water is 221 bar. When the reactor operates at 250 bar, the supercritical water does not exhibit a change in phase and the concept of boiling does not exist. The water density decreases continuously with temperature.

The inlet coolant temperature in lower part of the bed is 310 ºC and the outlet temperature in upper part of the bed is 416 ºC. Therefore, the water density decreases continuously from 0.725 to 0.137 g/cm^3 along the bed. This is an important factor to secure a more compact and better fixed core. The recommended pressure of 250 bar is due to the smooth and mild variation of density with pressure in this region resulting in the stability of flow in the core. The power production is much higher in this option as the difference in inlet and outlet enthalpy is much higher than in a conventional pressured or even boiling reactor. The plant thermal efficiency is estimated to exceed 40%. The turbines could be made smaller compared to those in conventional light water reactors. The superheated steam is fed directly into the turbine. Steam-water separation is not needed for direct cycle reactor. Some other advantages of such a choice include elimination of steam generators and reduced waste heat.

2.2. Fixed bed with helium gas as coolant

In this option, the fixed bed is cooled by helium yielding all the advantages of a gas-cooled reactor, including high efficiency and utilization of a direct gas turbine cycle. In this case we have a fast nuclear reactor system.

2.3. Fluidized bed with water as coolant

Power density may be significantly increased through fluidizing the bed. The increased turbulence of the coolant will allow a significant increase in power generation. In this case the effects of flow on the homogeneity of the fluidized bed porosity and on physical interaction between fuel elements need to be further studied.

2.4. Fluidized bed with supercritical steam as coolant

This option may take a further advantage of increasing the heat transfer rate, which would secure the production of a higher temperature steam and result in an even more efficient system.

3. SPECIAL FEATURES OF FBNR

The reactor is factory-fabricated and has no large or heavy components. The safety features of FBNR allow it to be built within or near urban areas. Long operation without on-site refuelling makes it appropriate for isolated remote places without infrastructure. FBNR could be the power source within a floating nuclear power plant. The reactor is equally appropriate for electricity generation, district heating, seawater desalination, process steam production or any combination thereof.

A combination of nuclear power reactor and seawater desalination plant could be realized in a more economical way, since the higher is the temperature and pressure of steam used in a turbine, the lower is the cost of electricity produced. On the other hand, steam at low temperature and pressure is needed for fractional desalination, and the greater part of inputted heat is the latent heat of steam. Therefore, the power production and desalination systems may be advantageously combined. A 100 MWe FBNR when realized within a cogeneration plant for the production of both power and potable water could produce 70 MWe of electricity and about 130,000 m^3/day of desalinated water.

4. MAJOR DESIGN AND OPERATING CHARACTERISTICS

The thermal power is about 30 MW per module. The power production depends on the power fraction allowed to feed the coolant pump. The reactor could be designed to produce two or three times more power. The reactor can operate in both base load and load-follow modes.

Two options are being considered for FBNR fuel elements. One is the zircaloy-cladded uranium dioxide spherical fuel pellet, and the other is spherical fuel element made of TRISO type coated particles. The enrichment for the UO_2/zircaloy option is about 3%, and for the coated particle option it is about 8%. Light water acts as both coolant and moderator. In a coated particle option, graphite also contributes to the neutron moderation. The module size will depend on fuel type and on the enrichment allowed. For example, the core of a reactor with UO_2/zircaloy fuel may have a diameter of 25 cm. The core tube may need to have larger diameter when coated particle fuel is used.

In the UO_2/zircaloy option, typical values of the reactor parameters are similar to those in conventional pressurized water reactors. FBNR is essentially a PWR with spherical fuel elements instead of the cylindrical ones. The moderator to fuel volume ratio in FBNR is about 0.8 in comparison to about 2 in conventional PWRs, which means that FBNR has a somewhat harder neutron spectrum.

The reactor makes no use of burnable poisons, and the reactivity margin is provided by fuel stored in the fuel chamber outside the core. For a slow reactivity insertion, the fuel is allowed to enter the core by lifting up the core height limiter located at the top of the core. The absorber shell is used for fine and prompt reactivity control.

The pressurized water-cooled reactor operating at 160 bar with inlet/outlet temperatures of 290 and 326 ^{o}C respectively will have an efficiency of about 33%, but the reactor cooled by supercritical steam may have an efficiency of more than 40%.

The high surface to volume ratio of spherical fuel elements ensures excellent heat transfer conditions, which results in low maximum and average fuel temperatures. In case of a coated particle fuel, the situation is even more favourable, since such fuel is designed to operate at very high temperatures. The coolant velocity is about 16 cm/sec. The core is cooled by forced convection, but the residual heat produced in the fuel chamber is removed by natural convection.

5. FUEL OPTIONS

As it was already mentioned, two options are being proposed for the FBNR fuel elements:

1. A spherical fuel element of 8 mm diameter made of uranium dioxide with the density of 10.5 g/cm^3, cladded by zircaloy;
2. A spherical fuel element of 8 mm diameter made of compacted micro-fuel-elements (MFEs) with the density of 5.9 g/cm3, cladded by silicon carbide.

MFEs are coated particles with the outer diameter of about 2 mm, similar to TRISO fuel. They consist of 1.5 - 1.64 mm diameter uranium dioxide kernels coated with 3 layers. The inner layer of 0.09 mm thickness, made of porous pyrolythic graphite (PyC) with the density of 1 g/cm^3, is called a buffer layer. It provides a space for gaseous fission products. The second layer of 0.02 mm thickness is a dense PyC (1.8 g/cm3), and the outer layer of 0.07 - 0.1 mm thickness is a corrosion resistant silicon carbide (SiC). The ceramic coating layers manufactured by chemical vapour deposition (CVD) method ensure the resistance of MFEs to water and steam at temperatures up to 950 ^{o}C and above. MFEs are also proved to confine fission products at temperatures up to 1600 ^{o}C over a long term.

In order to create an additional inhibition to power excursion, the isotopes of ^{175}Lu / ^{176}Lu or ^{181}Ta/^{182}Ta may be added to fuel. Harms et al. [13] argue that by introducing to fuel an isotope that contributes to large negative Doppler effect and has a large absorption cross-section, such as tantalum (^{181}Ta), the power excursion may be significantly eliminated. The more so as in case of a power excursion ^{181}Ta transforms into Ta^{182}, which in turn has a large neutron capture cross section and absorbs even more neutrons. This tandem effect should be further investigated.

6. TECHNICAL FEATURES AND TECHNOLOGICAL APPROACHES THAT ARE DEFINITIVE FOR REACTOR PERFORMANCE IN SPECIFIC AREAS

6.1. Economics and maintainability

The simplicity of design, short construction period and modularity result in a much smaller initial investment required for FBNR as compared to a conventional PWR. The provision of small units may match the energy needs as they arise, instead of providing large units to meet the uncertain energy requirements of the future. This could eliminate the necessity of a large initial investment and facilitate gradual rising of the needed funds.

The elimination of on-site refuelling and long operation time of the core will contribute to the reduction of operation and maintenance costs for FBNR. The absence of burnable poisons contributes to the improvement of neutron economy and therefore lower fuel enrichment will be needed. The simplicity of fuel fabrication for FBNR compared, for example, to the fabrication of fuel assemblies for a PWR may contribute to the reduction of fuel costs.

The inherent safety features of FBNR may make it possible to operate it totally by computers, thus avoiding human errors and reducing costs. Only an inspection and maintenance group may be required for periodical check-ups and maintenance of several reactors located at different sites.

The cost of electricity produced by FBNR is estimated to be around 0.02 $/kWh. Of this, the capital cost is about 0.016, the fuel cost is 0.003, and the operation cost is 0.002 $/kWh. The costs at such level compete well with the alternative energy sources. The total investment requirements to the design, construction and commissioning of an FBNR, including the investment during construction, are such that the necessary investment funds can easily be raised. The risk of investment in FBNR is sufficiently low compared to the risk of investment in other energy projects. Being small, factory-fabricated, having a construction period of only about 2 years, involving a relatively small investment of about 1000 $/kWe, and meeting the incremental character of increase in energy demand, FBNR could be viewed as an attractive long-term investment opportunity.

6.2. Provisions for sustainability, waste management and minimum adverse environmental impacts

The inherent safety features and small size of FBNR secure it impossible to produce a large release of radioactivity to the environment. However, envisaged as a defence-in-depth measure, simple underground containment will protect the environment against any possible adverse impacts.

FBNR is capable to utilize many types and combinations of fissile and fertile materials, such as uranium, plutonium, or thorium fuel. For example, plutonium from dismantled nuclear weapons could be used as fuel in combination with uranium or thorium, which is an extremely plentiful material in countries like Brazil and India. Therefore the fuel resources for FBNR are sufficient at least for centuries.

Spent nuclear fuel from FBNR is in a form and size that makes it possible to use it directly as a radiation source for irradiation purposes in agriculture and industry. It could also be reprocessed similar to LWR or HTGR fuel. Should reprocessing be not allowed, spent fuel elements could easily be vitrified in modules and the modules then be deposited directly to a waste repository.

6.3. Safety and reliability

The fuel elements in FBNR are suspended in the core by flow of coolant, and a passive cut-off of pump power will make them fall out of the core, driven by force of gravity, and go back into a sub-critical and passively cooled fuel chamber.

The FBNR fuel appears to be more robust than that of a PWR. FBNR core appears as a suspended set of loosely packed spheres, whereas PWR fuel assemblies are complex and delicate structures involving grids and thimble rods. In particular, no seismic impact on FBNR fuel is predicted.

Each FBNR fuel element contains less than 0.25% of the fuel present within a single fuel rod of conventional PWR and operates at a significantly lower temperature. Therefore any failure of such fuel elements will produce significantly smaller consequences. It is very difficult to imagine a scenario in which FBNR fuel failure can cause a major release of radioactivity, therefore the need to plan human relocation or evacuation measures beyond the plant boundary could be eliminated.

FBNR makes use of the PWR technology, for which safety design basis has already been well established through the efforts of the IAEA and many national nuclear organizations. The computer codes used to analyse FBNR are the modified versions of the codes used in the design and analysis of conventional pressurized water reactors. Here, only the model of equivalence between a spherical and a cylindrical fuel element needs to be verified. Verification could be provided through tests performed on a single FBNR module, which consists of a pump and a 25 cm diameter tube that is partially filled with spherical fuel pellets. Such module could be accommodated within a standard PWR test facility. The prototype of FBNR would consist of a single module that can easily be built at relatively low cost.

Since FBNR fuel appears as a set of small spherical fuel elements, its transportation could be much more simple than that of ~4 m long PWR fuel assemblies. The decommissioning of FBNR could also be a relatively simple job, since the reactor is modular in design and the modules are small in size and weight. There is no heavy pressure vessel in FBNR.

6.4. Proliferation resistance

The irradiated coated particle fuel discharged from the reactor is very resistant against heat and nitric acid. It is rather difficult to reprocess it. Special mechanical treatment is required and this technology is not widely available. Therefore, the reactor grade plutonium contained in spent FBNR MFEs is less accessible than plutonium from standard LWR spent fuel.

Adopting thorium fuel cycle may be an intrinsic measure to hinder the possibility of misuse of nuclear materials for a weapon programme. Mixing of thorium with low enriched uranium results in the production of U-233 that is diluted with U-238. The access to pure U-233 will only be possible through a sophisticated isotope separation technique. In thorium cycle, high Pu-238 to Pu-239 ratio and the production of gamma emitting Tl-208 are hindrances to nuclear proliferation.

6.5. Technical features and technological approaches used to facilitate physical protection

FBNR fuel is contained in a sealed module and is hard to access. The fuel remains in the core only under the conditions of normal operation when the reactor is critical. In any other situations the fuel leaves the core and remains in a sub-critical state under a passive mode of cooling. No scenario where sabotage can provoke severe adverse consequences is envisaged.

6.6. Non-technical factors and arrangements that could facilitate effective development and deployment of reactor installation

FBNR may have a variety of applications, including electricity generation, production of steam for industrial purposes, seawater desalination, or a combination thereof. Also, FBNR could rely on the existing nuclear power infrastructure and legal institutions.

To facilitate the development and deployment of FBNR it is desirable that a new institution, World Nuclear Energy Company (WONEC) is formed to become a catalyst in organizing and coordinating the scientific and industrial potential of many countries for the provision of safe and clean energy supply to the world. This company could act as a commercial as well as scientific venture, with its shares being freely traded in the international financial market. WONEC could eventually supply FBNR and be responsible for its entire fuel cycle. The project is to remain a totally scientific, industrial, and economic venture, avoiding dominating national politics, and be in conformity with the spirit of the new age and the presently growing international desire for world peace.

The developing countries are expected to show great interest to participate in WONEC, since in this way they will acquire nuclear power without the fear of being exploited by vendors or making very large investments for an independent national nuclear programme. The industrialized countries are expected to support the idea as well, since by participation in WONEC they will benefit from the sales of their technologies to WONEC and partake in a very large nuclear reactor sales market worldwide.

WONEC, an international consortium responsible for the development and deployment of FBNR, could operate closely and faithfully with the IAEA to control the problem of nuclear proliferation.

7. ENABLING TECHNOLOGIES RELEVANT FOR FBNR

For a coated particle fuel option, there is a need to develop the technology to manufacture 8 mm spherical fuel elements from coated particles. These fuel elements should then be tested under irradiation. Since the irradiation involves small quantity of materials, it can easily be performed at a number or research reactors available around the world.

For spherical uranium dioxide pellets, the technology of fabrication of 8 mm spheres from UO_2 powder has to be developed. As a first step, grinding of the cylindrical PWR type pellets could be tried to produce spheres. The technology of pressing of zircaloy hemispheres that could be then welded together with a spherical fuel pellet put in between could be developed to produce complete fuel elements. The testing procedures are expected to be similar to those used in PWR fuel fabrication.

8. STATUS OF R&D AND PLANNED SCHEDULE

R&D programme

The currently available R&D programme for FBNR visualizes the following steps:

- Development of conceptual designs for selected fuel options;
- Construction of a full size non-nuclear hydraulic module to verify hydraulic performance of the core and primary circuit and to determine basic thermal-hydraulic correlations;
- Performance of neutron-physical, thermal hydraulic and structural calculations;
- Adjustment of codes and performance of safety analysis;
- Elaboration of fuel fabrication technology;
- Fabrication and testing of fuel samples for irradiation testing;
- Performance of fuel testing under irradiation and of post irradiation examinations;
- Development of an engineering design of FBNR prototype;
- Performance of zero power experiment with a single-module at a critical facility;
- Construction of a single-module prototype.

Hydraulic study of the fixed bed suspended core concept

Hydraulic studies of a full size suspended mock-up core could be performed at an experimental hydraulic module made of transparent materials and using stainless steel balls to simulate fuel elements. Such module is to be provided with instrumentation to measure basic hydraulics parameters such as pressure drop as a function of coolant flow, velocity under different core configurations, etc. Videotaping of core operation could be provided in order to analyze performance of the suspended core under various simulated operating and accidental conditions.

9. OTHER SIMILAR OR RELEVANT REACTOR CONCEPTS FOR WHICH THE DESIGN ACTIVITIES ARE ON-GOING

All-Russian Nuclear Machinery Institute (VNIIAM) and Russian Research Centre “Kurchatov Institute” in Moscow, Russian Federation, and the Pacific Northwest National Laboratory (PNNL) in the U.S.A. perform joint R&D activities for the development of a 300 MWe pebble bed boiling water reactor without on-site refuelling and of a 1500 MWe pebble bed direct flow reactor [14].

10. CONCLUSION

A concept of Fixed Bed Nuclear Reactor (FBNR) is proposed, in which spherical fuel elements are kept at a fixed position within the upper part of the core by light water coolant flow. Passive cut-off of pump power under any emergency conditions makes the fuel elements fall out of the core, driven by force of gravity, and go down into a sub-critical and passively cooled fuel chamber. FBNR appears as a simplified version of the well-known fluidized bed nuclear reactor concept proposed earlier. FBNR is proposed for lifetime core operation without on-site refuelling. The preliminary consideration has shown that simplicity and passive safety characteristics of FBNR together with the reliance on a well-established PWR technology may make it a viable option for the near future deployment. A programme of R&D is presented that, due to the small size and weight characteristics of a 30 MWe module, could be implemented at many existing experimental facilities around the world.

Among others, the FBNR concept was considered at the IAEA consultancy meeting on Small Reactors without On-site Refuelling in March 2004, attended by experts from 6 IAEA Member States and the comment on FBNR was as follows: "In particular, the consultancy noted that the innovative approach proposed in the Fixed Bed Nuclear Reactor (FNBR) concept (Federal University of Rio Grande Do Sul, Brazil) relating to hydraulically supported column of spherical fuel elements offers a good potential to serve as a method of passive control of core reactivity. This concept needs to be further developed and polished for its possible implementation in small reactors."

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ANNEX 13

APPLICATION OF "CANDLE" BURNUP TO LBE COOLED FAST REACTOR

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Abstract. A new reactor burnup strategy CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor) was proposed [1], where shapes of neutron flux, nuclide densities and power density distributions remain constant but move along axial direction of the core. The equilibrium state was obtained for a large fast reactor (core radius of 2 m) by using a newly developed direct analysis code. Only natural or depleted uranium is enough to be charged to the core starting from its second sequential lifetime. In other words, if the fuel for the first core is available, neither enrichment nor reprocessing are required further on. In this, the burnup of spent fuel is about 40%. However, it is difficult to apply this burnup strategy to small reactors (core radius of ~0.8 m). A long-life small fast reactor with nitride fuel and Lead Bismuth Eutectics (LBE) coolant was investigated. In was shown that enriching fuel by 5% of plutonium makes it possible to realize CANDLE burnup for small reactors. The core of 2 to 3 m height can have an operation period of 9 to 28 years with very small reactivity change (less than 0.2%). Though the application of CANDLE burnup strategy to fast reactors will require many changes in their design, CANDLE application to block-type fuel hight temperture gas cooled reactor seems to require only few design changes.

1. INTRODUCTION

1.1. CANDLE Burnup Strategy

1.1.1. What is CANDLE burnup?

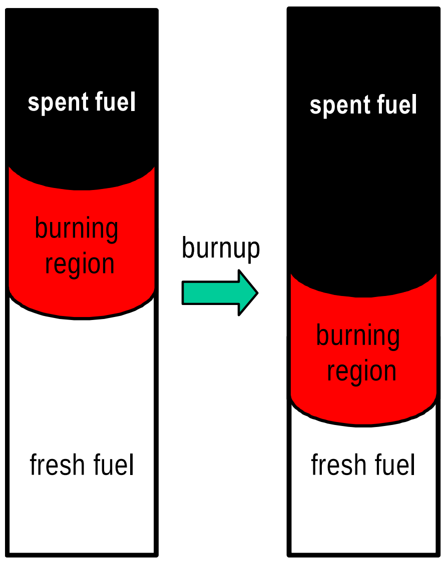

FIG. 1. Concept of CANDLE burnup strategy.

A new reactor burnup strategy CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor) is proposed [1,2], where neutron flux, nuclide densities and power density do not change their shapes along burnup but move in the axial direction of a core with a constant velocity under constant power operation, as shown in Fig. 1. The moving direction can be chosen either upward or downward, but in the present paper only the downward direction is considered for making description of the paper simple.

The CANDLE burnup strategy can be applied to several reactors, for which the infinite neutron multiplication factor of a fuel element changes along burnup in a proper way, such as it changes from less than or nearly unity to considerably more than unity, and then to less than or nearly unity again. When this condition is satisfied, in the lower part of the burning region the infinite neutron multiplication factor increases with burnup, and in the upper part of the burning region it decreases with burnup. Therefore, the burning region shifts downward. Both natural uranium and depleted uranium show this behavior of their infinite neutron multiplication factors in fast neutron field with hard neutron spectrum. In the present paper only fast reactor cases are studied, even though a thermal reactor can offer interesting examples of CANDLE burnup by introducing highly enriched fuel and burnable poisons with high microscopic absorption cross section.

The axial distribution of each nuclide density in fresh fuel region is uniform, but it is more complex in the burning region. It may be difficult to construct the ignition region for this burnup strategy. However, the set-up of the succeeding core configuration is very easy. The burning region at the end of reactor life can be used as the ignition region for the succeeding core as shown in Fig. 2.

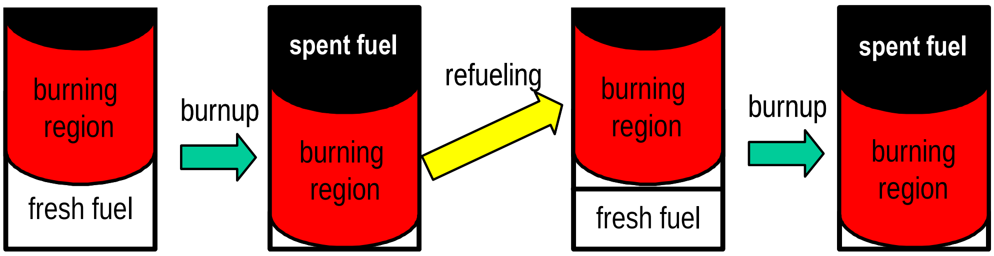

FIG. 2. Concept of CANDLE burnup and refuelling strategy.

1.1.2. General merits and demerits

The general merits of CANDLE burnup are as follows:
(a) Burnup reactivity control mechanism is not required.

The benefits resulting from this are:

- the reactor control becomes simpler and easier;
- the excess burnup reactivity becomes zero, and the reactor becomes free from reactivity-induced accidents;
- the burnup of control rods becomes negligible, as neutrons are not absorbed by absorber but efficiently utilized.

(b) The reactor characteristics such as power peaking and reactivity coefficients do not change with burnup.

The benefits resulting from this are:

- the expectation of core condition becomes very reliable;
- the reactor operation strategy remains the same for different burnup stages and, therefore, the reactor operation becomes simple and easy.

(c) Orifice control is not required under changing fuel burnup.

Since the radial power profile does not change with burnup, the required flow rate for each coolant channel does not change. There is no need in orifice control and, therefore, relevant operational errors could be avoided.

(d) Radial power distribution can be optimized more thoroughly.

Since the radial power distribution does not change with time, it can be optimized more thoroughly. The optimization method becomes much more simple. By choosing properly the radial fresh fuel distribution, the discharged fuel burnup can be equalized for every fuel discharge. At the same time the integral power output for each radial position becomes equalized, as well as the coolant temperature at core outlet for each coolant channel in the core.

(e) Reactor core life may be elongated.

By simply increasing the core height the reactor life can be elongated. Therefore, design of a long-life core reactor becomes easier.

(f) Recriticality accident caused by CDA is avoided.

Since the reactor is just critical without any absorbers and contains no surplus fissile materials in its core, recriticality accident caused by CDA could hardly happen.

(g) Infinite neutron multiplication factor of fresh fuel is less than unity.

Therefore, the risk for criticality accident is small, and the transportation and storage of fresh fuel becomes simple and safe.

At the same time CANDLE burnup may have the following demerits:
(a) Coolant pressure drop becomes larger, since the core becomes higher;

(b) There is less freedom for optimization of core axial power distribution.

However, these demerits could not be considered fatal because core height is a function of axial power profile and the speed of the burning region drift. Both of these values may be made small for many designs, and some examples will be provided later on in the paper. Therefore, the demerit (a) is not essentially important. As comes to the demerit (b), the total power distribution can be optimized to a higher level for this burnup strategy, since the radial power distribution could be optimized very thoroughly.

1.1.3. Application to fast reactors

In this paper the applications of CANDLE burnup strategy to fast reactors are discussed. The infinite neutron multiplication factor of natural uranium in fast neutron spectrum satisfies the required condition mentioned above. However, the excess reactivity is marginal. Though a hard spectrum large fast reactor can realize CANDLE burnup strategy, most small fast reactors cannot realize it with only natural uranium because of the large neutron leakage from their cores.

The above-mentioned characteristics are general characteristics of CANDLE burnup strategy for fast reactors. Mentioned below are outstandingly good characteristics that could be obtained when CANDLE is applied to fast reactors with excellent neutron economy [1,2]:

(a) Enriched fuel is not required after the second core.

Only natural or depleted uranium is enough to be charged to the core starting from its second sequential lifetime. In other words, if the fuel for the first core is available, neither enrichment nor reprocessing are required further on.

(b) The burnup of the spent fuel is about 40%.

This value is competitive to the value of the presently projected fast reactor systems with reprocessing of spent fuel. In CANDLE, 40% of natural uranium is burned without enrichment or reprocessing.

(c) Long-life reactor can be designed easily, since the burning region drift speed is only about 4 cm/year for the conventional power density level.

Even a reactor with a core lifetime of 30 years can be designed simply by adding 1.2 m to the initial core height.

The problems for this case are as follows:

(a) The reactor is required to pertain very high neutron economy.

However, many designs satisfying this requirement may be presented.

(b) The fuel material should pertain integrity performance for very high burnups.

Though there are some experimental results for such a high burnup, their number is insuficient at present.

The merits for this case may be outstanding, but the problem for small reactors appears to be very difficult. The criticality requirement can be satisfied by adding some plutonium to fresh fuel of a small reactor. However, the requirement that infinite neutron multiplication factor of fresh fuel should be less than or nearly unity is difficult to be met. On the other hand, strict implementation of CANDLE burnup strategy may be not needed for these reactors, once several other important characteristics are ensured, such as long-life core, simplicity and inherent safety. In this paper the application of CANDLE burnup strategy to small fast reactor with plutonium-admixed natural uranium as fresh fuel is investigated.

In the studies presented only natural uranium was treated, but depleted uranium can also be utilized for the same purpose., the more so as there is a large amount of depleted uranium being stored worldwide at present. However, to make the story simple, only natural uranium is discussed in this paper.

Long-life small reactors are considered very important to solve present global energy and environmental problems. Before descussing the application of CANDLE to such reactors, a discussion on long-life small reactors will first be presented in the next section.

1.2. Long-life small reactor

1.2.1. Problems for the past trend of NP reactors and the potential of small reactors [3]

The size of conventional nuclear power reactors has almost reached its maximum limit, after pursuing their economy by exploiting scale merit. It may be difficult to find a future way for them to be deployed. Soon it becomes almost impossible to find any new site for them in the developed countries. They also have a large economical risk unbearable even for large companies or even governments.

Small reactors can be built on a less graded land, such as small and less stable land. Therefore, it is much easier to find sits for small reactors.

Nuclear reactors can also be utilized for several purposes other than electricity production, such as heat generation, seawater desalination, hydrogen production, etc. Since the transportation of heat, potable water or hydrogen for long distances requires high costs and faces energy and material losses, small reactors providing the flixibility in siting become preferable.

Present power-plant owners hesitate to build a new NPP with larger reactor for the reason of large associated economical risk. Even a delay in construction may cause considerable economical damage. They may prefer to build a smaller reactor if it is economically feasible.

The scale demerit is a considerable factor to degrade the economical performance of all small reactors. However, there are many items pertaining to small reactors that may improve their economical performance. Some small reactors can be built in a factory and this can reduce the reactor cost considerably. Since the number of required reactors for a given power rate becomes larger when several small reactor instead of one large reactor, more experience and learning could be obtained for systems with small reactors. The term required for licensing and construction may be made shorter for small reactors, and the amount of interest on investments becomes also smaller. Modular systems based on small reactors are expected to work efficiently with excellent economical performance within NPPs of large overall capacity.

1.2.2. Small reactors for developing countries

It is well known that smaller reactor is safer than the larger reactor, since its systems are simpler and the total amount of contained radioactive material is smaller. Furthermore, certain small reactors possess inherent safety characteristics, where safety function relies more on natural phenomena and less on human action or mechanical devices.

If a reactor is transportable and has a long life, it can be built in a factory and shipped to the site and installed and operated there within a certain period of time without refuelling. After its lifetime is completed it could be shipped back to the factory and replaced with a new reactor. When the vessel of such reactor is designed to be sealed, this reactor may possess excellent proliferation resistance features.

In the 21st century global warming caused by the carbon dioxide emissions becomes an inevitable problem. Especially, the carbon dioxide emissions from developing countries may become important. Nuclear reactors are free from this problem.

However, many developing countries do not have the sufficient infrastructure and number of trained personnel. Furthermore, some developing countries are politically unstable. The energy demand in these cases is usually local and small. As already mentioned, small reactor may be made easy for operation and maintenance. It may also be made inherently safe and proliferation resistant. Therefore, small reactors have a large potential to solve global warming problem.

The target reactor for this purpose should be transportable, long-life, safe, simple (easy maintenance and operation), and proliferation-resistant. However, some of these characteristics are tightly related to each other. By investigating these characteristics, it appears that only long-life core and small-size are basic characteristics and that other characteristics can be derived from these two.

1.2.3. LBE cooled fast reactor

In the above discussion it appears that long-life and small-size are basic requirements for realizing long-life, safe, simple, small, transportable, and proliferation-resistant reactor. However, the approaches to make reactor long-life and small-size are in conflict because a small-sized reactor usually shows poor neutron economy, hence higher burnup can not be expected. The neutron economy, namely reactor criticality, limits both size and life of the reactor. Our discussion leads to the conclusion that long-life, safe, simple, small, transportable, and proliferation-resistant reactor requires excellent neutron economy. It is well known that fast reactors show much better neutron economy compared to thermal or epithermal reactors.

The above discussion suggests that we should investigate small fast reactors. At present, sodium is considered as the best coolant for fast reactors. The main reason is its superior cooling ability. It facilitates high power density and makes the doubling time shorter. Short doubling time was an indispensable requirement in the early time of fast breeder reactor history. It is reported that from a safety viewpoint lead-bismuth-eutectic (LBE) was originally considered [4].

As mentioned before, neutron economy is very important for realizing long-life small reactor. For the small fast reactor, it is expected that LBE coolant shows much better performance on the neutron economy than sodium coolant because of its large scattering cross section and heavy nuclide mass. It is reported that LBE-cooled long-life small fast reactor shows better performances for neutron economy, burnup reactivity swing and void coefficient [3,5]. However, in the western world it has been accepted for a long time that LBE cannot be used as a reactor coolant because of the associated corrosion problem. In Russia this problem was solved by oxygen concentration control, and LBE was employed as a submarine reactor coolant. It is reported that 8 nuclear submarines with LBE coolant were constructed and operated for about 80 reactor-years [4]. After the results of Russian research were declassified, many studies of LBE, especially corrosion tests, were initiated worldwide. The corrosion problem could be solved by the selection of proper structural material, temperature, fluid velocity and oxygen concentration.

1.2.4. Characteristics of LBE

The most important merit of LBE compared to sodium is chemical inertness. LBE does not react violently with water or air.

The boiling point of sodium is 1156 K, and it is not easy to prevent its boiling in some severe accidents. If the void coefficient is positive, an accident may lead to core destruction. The boiling temperature of LBE is 1943 K, and the possibility of boiling seems negligible. Furthermore, the void coefficient is more negative than in sodium case, as mentioned before.

The density of LBE is about 12 times of the sodium density. The viscosity of LBE is large and, therefore, pressure drop is expected to be large. The Prandtl number is about 3 times of the value for sodium. These characteristics lead to the poor cooling ability of LBE, and then the power density of LBE-cooled reactor should be lower. Anyway, from reasons of corrosion protection the flow speed for LBE has to be set lower.

Power density of small reactors is usually restricted by the minimum size of their cores as defined by criticality condition. For some very small fast reactors power desnity should be kept very low even if sodium coolant is used. Therefore, poor cooling ability of LBE is on itself not a big disadvantage for long-life small reactors.

For natural circulation capability, LBE-cooled reactors can offer a better potential provided by larger equivalent hydraulic diameter of their cores. It also improves the response of a reactor in accidents.

As mentioned before, the LBE-cooled long-life small fast reactor shows better performances for neutron economy, burnup reactivity swing, and void coefficient. The LBE also secures noticeable shielding for neutrons and gamma-rays. Then, the reactor size can be reduced.

Radioactive materials produced in the coolant during operation are also important. For sodium ^{24}Na should be considered. Its half-life is 15 hours, and it emits high-energy photons (2.8 and 1.4 MeV). Therefore, the primary loop of sodium cooled reactor is a source of very high dose rate. On the contrary, LBE does not produce much gamma-ray emitters, though Polonium is produced which is an alpha- emitter. Then, the dose-rate around the primary loop of LBE resactor is expected to be much lower than in the sodium case.

2. CALCULATION METHOD

The presented work is a first-trial application of the CANDLE burnup strategy to small fast reactors. To begin with, the equilibrium state satisfying the CANDLE burnup requirements was tried to be obtained with the use of a steady state code [2]. Once the shape is obtained as realistic value for a given reactor design, the simulations of such reactor are performed as the next and final step.

To calculate steady state CANDLE burnup, Galilean transformation is used in order to set the burning region at rest in the transformed coordinate system. Otherwise, the calculation region is expanding, since the burning region moves steadily with each next iteration of the calculation. The convergence judgment becomes quite easy for this coordinate system. The actual mathematical treatment and calculation method are described in reference [2].

The follow-up reactor simulations were performed using standard neutron transport – fuel burnup procedures.

3. CALCULATION CONDITIONS

Though the subject of this work is the application of CANDLE burnup strategy to a small fast reactor, typical application of CANDLE to a large fast reactor with natural uranium feed is also presented as reference. The design parameters of large and small fast reactors are shown in Table 1.

It is difficult to keep criticality for a small reactor, since its neturon leakage is larger and its neutorn economy is more poor than in a large reactor. In the present study this was compensated by adding some plutonium to the fresh fuel of a small reactor. However, this method being applied, the requirement to fresh fuel, which is that its infinite neutron multiplication factor should be less than or nearly unity, becomes difficult to be satisfied. However, strict implementation of CANDLE burnup strategy may be not needed for the considered small reactor, since several other important characteristics are ensured, such as long-life core, small core size, and small burnup reactivity swing.

TABLE 1. DESIGN PARAMETERS FOR LARGE AND SMALL FAST REACTORS

thermal output		3000 MWth	300 MWth
core radius		2.0 m	0.8 m
core height		8 m (infinity)	2.0, 2,5, 3.0 m
radial reflector thickness		0.5 m	0.5 m
material	fuel	$U_{0.77}Zr_{0.23}$	(U, Pu) N
	cladding	HT-9	HT-9
	coolant	Pb-Bi (44.5,55.5%)	Pb-Bi (44.5,55.5%)
cell type		fuel pin	tube in shell
fuel pin diameter		0.8 cm	
coolant channel diameter			0.668 cm
cladding thickness		0.035 cm	0.035 cm
fuel	theoretical density	15.90 g/cm^3	14.32 g/cm^3
	fuel smear density	75%	80%
fuel volume ratio		50%	61%

Metallic fuel, which secures excellent neutron economy, was chosen as a fuel material for the large reactor, while for the small reactor a more practical nitride fuel was used. In order to improve neutron economy for small reactor, its fuel volume ratio was increased by employing the tube-in-shell type fuel cell. For large reactor standard pin-type fuel was used.

Cylindrically symmetric model of a reactor was considered. The core height of large reactor was set to be 8 m, which is equivlent to infinity for neutron transport calculations. On the other hand, more practical core height values of 2.0, 2.5 and 3.0 m were set for small reactor. The core height should be determined by considering the axial power distribution and the burning region speed as defined by the CANDLE burnup strategy. The average power density of large reactor is expected to be 1.6 times higher than for small reactor, if the core height is the same for both reactors. Since small reactor generally targets inherently safe performance, its power density is usually set lower.

The neutron transport was treated by a 21-group diffusion equation, where the group constants and their changes with respect to temperature and atomic density were calculated using a part of the SRAC code system [6] with JENDL-3.2 nuclear data library [7]. Twenty actinides and 66 fission products were treated in the calculations.

4. CALCULATION RESULTS

The results of steady-state calculations performed under the conditions mentioned above are shown in Table 2.

TABLE 2. CALCULATION RESULTS

thermal output	3000 MWth	300 MWth		
fuel material	metal	nitride		
Pu enrichment	0%	5.0%		
core height	infinity	3.0 m	2.5 m	2.0 m
k_{eff}	1.003	1.010	1.009	1.004
shift speed of burning region	4.25 cm/year	2.67 cm/year	2.71 cm/year	2.73 cm/year
average burn up of discharged fuel	38.20%	31.30%	30.9%	30.30%

The value of the effective neutron multiplication factor, k_{eff}, should be set to unity, since this calculation is performed for the equilibrium condition at operation temperature. However, since this design is at a preliminary stage, it was set to be more than unity providing for a small uncertainty margin. As it was already mentioned, the obtained average burnup of the discharged fuel is very large. It presents not only a very good fuel cycle performance mentioned before, but also poses problems associating with the performance of materials. The spent fuel burnup of a small reactor is lower than for a large reactor, but is still much higher than in a conventional reactor. The shift speed of burning region is small enough for a long-life reactor.

TABLE 3. SIMULATION RESULTS FOR SMALL REACTORS

Core height	3.0 m	2.5 m	2.0 m
Operation period	28 years	18 years	9 years
Maximum change of excess reactivity during cycle (dk/k)	0.16%	0.14%	0.14%

The simulation studies of CANDLE burnup were performed for several core heights. The results obtained are shown in Table 3. The operation period is seen increasing when the core height is increased, an acceptable value of the excess reactivity change during cycle being kept. Figure 3 shows more vividly the relationship between core heignt and operation period. The operation period observed is long enough for a long-life reactor. For a given core life, the core height can be estimated with good accuracy as (power shape width) + (shift speed of burning region) x (core life). Increasing core height in CANDLE appears to be a much more effective approach to the increase of core lifetime as compared to the traditional reduction of power density, when core height can be estimated as (standard core height) x (design core life) / (standard core life).

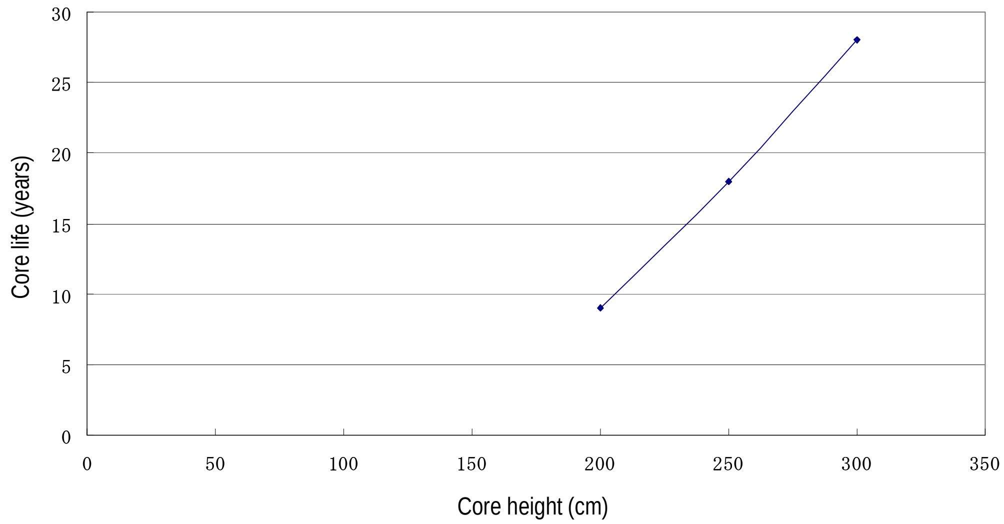

FIG. 3. Relation between core height and core life for CANDLE burnup.

5. CONCLUSION

A new reactor burnup strategy CANDLE was proposed, in which relative distribution shapes of neutron flux, nuclide densities and power density are constant over reactor lifetime but move in the axial direction with a velocity proportional to the power rate during the whole life of reactor operation. This strategy requires a special pattern of the core's infinite neutron multiplication factor changes with burnup, and can be applied to a fast reactor with good neutron economy.

The equilibrium state was successfully obtained for a large fast reactor with the core radius of 2 m. Only natural or depleted uranium is enough to be charged to the core starting from its second sequential lifetime. In other words, if the fuel for the first core is available, neither enrichment nor reprocessing are required further on. The burnup of spent fuel is about 40%.

For a long-life small fast reactor (core radius 0.8 m), enriching fuel by 5% of plutonium makes it possible to realize CANDLE burnup. The core height of 2 to 3 m can secure the operation period of 9 to 28 years with a very small excess reactivity change (less than 0.2%).

Though the application of CANDLE burnup strategy for fast reactors requires a lot of changes in their design, the application of CANDLE to block-type fuel hight temperture gas cooled reactor seems relatively easy [8].

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